F.2.1 Chemical Compatibility

The most important criterion in assessing foreign research reactor spent nuclear fuel storage technologies is the compatibility of the spent nuclear fuel with the fuel storage technology environment. The research reactor fuel cladding is either aluminum or stainless steel. Aluminum cladding fuel is the predominant type in the mix of foreign research reactor spent nuclear fuel being considered for acceptance in this EIS. The selected method of storage for any foreign research reactor spent nuclear fuel that may be accepted must provide a benign and noncorrosive environment for the fuel.

In reviewing the corrosive potential of aluminum, acidic, alkaline, and even many neutral chemical solutions have been found to be significantly corrosive. Therefore, the use of wet storage technology for the majority of the foreign research reactor spent nuclear fuel that contains aluminum cladding would require the maintenance of high water purity throughout the storage life of the pool, which may equal or exceed 40 years.

Unlike wet storage, most of the dry storage technologies utilize a dry inert gas atmosphere for the fuel, which is a noncorrosive noble gas that also enhances conduction heat transfer from the fuel to the encapsulating container. Some dry storage technologies use dry nitrogen instead of inert gas. Even in the event of a loss of inert gas atmosphere, the air atmosphere would be less corrosive than a less-than-high-purity water pool. Finally, previous experience at many DOE wet storage facilities has shown that poor water quality dramatically deteriorates the integrity of aluminum fuel. Thus, the chemical compatibility criterion indicates that dry storage is a more appropriate technology than wet storage for extended storage of foreign research reactor spent nuclear fuel.

F.2.2 Subcriticality Assurance

Uranium and plutonium are the principal elements that have the unique ability to split or fission after absorbing a neutron, and release energy and several new neutrons from this fission process. The particular forms or isotopes of uranium that are effective in the fission process are called fissile materials and are ^{233}U, ^{235}U, and ^{239}Pu. Of these three isotopes, only ^{235}U exists naturally, while the other two isotopes can be produced artificially. Under the right conditions, the fission process can be self-sustaining or even grow by a chain reaction. This chain reaction produces as many or more neutrons than are absorbed in an assembly of fissile materials.

In nuclear engineering terminology, the numerical measure of a mass of fissile material to achieve and maintain a self-sustaining fission chain reaction is termed K-effective. K-effective is the net ratio of neutrons produced per neutron absorbed in the fissile material mass. When K-effective equals 1.0, the mass is said to be critical because it can maintain the fission process. When K-effective is less than 1.0, the mass is considered to be subcritical.

Subcriticality can be ensured by a number of factors, including:

- diluted concentration of fissile materials,
- adequate separation distance between masses of fissile materials such as nuclear fuel rods or assemblies,
- presence of materials (such as boron) mixed with the fissile material that absorbs neutrons before they can be captured by the fissile material,

- exclusion of substances such as water that can encourage the absorption of neutrons by the fissile isotopes or reflect neutrons leaving the mass of fissile materials back into the mass, and

- restricting the mass of fissile material below the minimum that nature requires to initiate, maintain, and/or sustain a fission chain reaction.

In nuclear criticality safety, the principle of double contingency is used to protect against criticality. Double contingency requires that the design of any system containing fissile material use two of the aforementioned factors to prevent the onset of criticality. Criticality analyses would be required to confirm spacing and the effects of optimum moderation, as well as the different structural materials in foreign research reactor spent nuclear fuel as compared to commercial fuel (e.g., aluminum, stainless steel, hydride, inconel in foreign research reactor spent nuclear fuel as compared to zircaloy in commercial fuel).

Another important factor is that most of the products of the fission reaction are radioactive fission products that are not capable of sustaining a fission reaction. Like most elements in nature, these fission products can absorb neutrons, but do not produce any neutrons or energy during this absorption. Mixed with these fission products is a small amount of fissile ^{239}Pu, which was also created during the fission process. However, the sum of the remaining ^{235}U and the created ^{239}Pu is much smaller than the original quantity and concentration of ^{235}U in the new fuel. The relatively low (in comparison to its initial value) ^{235}U enrichment and ^{239}Pu present in foreign research reactor spent nuclear fuel, coupled with the presence of neutron-absorbing fission products, greatly reduces the physical ability of foreign research reactor spent nuclear fuel to become critical.

Another important aspect in selecting an appropriate storage technology is the maintenance of subcriticality. For wet (pool) storage, subcriticality is ensured by fuel spacing, and in some cases, the use of spacing plates between adjacent fuel assemblies that contain boron. In addition, control of the maximum allowable concentration of fissile isotopes (i.e., ^{235}U enrichment) is another method used to control subcriticality. Since all dry storage technologies use a storage canister for fuel, the subcriticality design relies on controlling the fissile material inventory, fuel spacing, and if necessary, the use of neutron-absorbing materials. The subcriticality control design of all fuel storage technologies is acceptable and does not provide any discriminating factors for selecting one technology over another.

F.2.3 Shielding Effectiveness

Shielding effectiveness design impacts both onsite worker and public dose rates during the loading and subsequent storage of spent nuclear fuel. Both neutron and gamma ray shielding must be provided and ensured throughout the life of the storage facility. Wet storage technology uses pool water as a shield, which effectively reduces both neutron and gamma doses to acceptable levels. The only weakness of this shielding design is any event in which the water could be lost due to a leak in the pool wall or, if the design includes piping at a low enough level, a pipe break in a line connected to the pool for the water purification or decay heat removal systems. Precluding piping or wall leakage, spent nuclear fuel pool water is an inexpensive shield medium that offers the advantage of providing visual inspection of the stored fuel as long as water purity and clarity are maintained.

Dry storage technology relies on a number of solid shielding materials, sometimes in combination, to reduce gamma and neutron dose rates. The most common materials are different forms of concrete (i.e., low-density, high-density, hydrogenated), cast iron, carbon or stainless steel, lead, borated resin, and polyethylene (for neutrons). As with water, these materials have been widely used in the nuclear industry for shielding and their properties are well known. They are more costly than pool water and prevent visual inspection of the spent nuclear fuel, but are not prone to material loss like pool water.

In comparing shielding designs, it is important to note that most of the shielding materials have inherent limiting temperatures (i.e., maximum allowable temperature) with the exception of the steels and cast iron. These metals' temperature limits are much greater than the aluminum-based fuel cladding temperature limit. Shielding material thermal limits include both absolute values of temperature and, in the case of concrete, temperature gradients that create thermal stresses. Wet storage pool water also has a thermal limit that is the prevention of local or bulk boiling in the pool. Operation of the spent nuclear fuel pool heat removal system prevents pool water boiling, but a postulated accident in which this system is disabled requires calculation of the time before the inception of bulk pool boiling. Adequate natural convection between adjacent fuel assemblies and within storage racks prevents local nucleate boiling in any fuel flow channel.

Shielding geometry plays an important role in the determination of a dose rate profile around the storage facility. A continuous and constant thickness of shielding completely surrounding the fuel provides a relatively constant dose rate at all locations. A shielding design that is asymmetric and contains air gaps and/or varying material thickness results in hot spots and a relatively larger variation in surface dose rates. The wet pool design offers a continuous shield of water with resulting low constant dose rates throughout the pool surface. The water and pool wall, usually steel-lined concrete, also maintain a low continuous dose rate profile outside the walls.

The dry storage concrete building and concrete cask technologies rely on concrete walls for shielding with some steel internal to the walls. The need for internal airflow passages in the concrete introduces gaps in these walls. These gaps, which are labyrinths, require complex shielding analyses and typically allow a relatively larger dose rate at the air inlets and/or outlets than at the bulk concrete wall. This effect is more significant for concrete casks than for concrete buildings because the casks are more limited in the concrete thickness that is used in their shield wall. In the case of metal casks, since there are no internal air passages in the metal shield, the dose rate is relatively uniform around the surface. Different axial shielding and neutron-gamma source terms will result in different axial dose rates for the metal and concrete casks. Inground storage systems use a relatively small amount of concrete radially coupled with the surrounding earth for shielding and employ thick steel plugs for axial shielding. The hybrid metal-concrete cask design uses shielding principles similar to the concrete cask.

In comparing spent nuclear fuel storage shielding designs, the four basic technologies can be characterized as water, lead, metal, and concrete. Water and metal provide the most uniform dose rate reduction because they do not require the inclusion of labyrinth airflow passages for decay heat removal necessary for concrete. Water is most susceptible to a sudden rapid loss of shielding effectiveness because it is a liquid requiring confinement. It should be noted, however, that pool storage of spent nuclear fuel has been effectively used in the nuclear industry for over 40 years. Both concrete and water are susceptible to degradation of their shielding effectiveness if temperature limits are exceeded. These thermal limits can be accommodated by proper design of the spent nuclear fuel pool cooling system for water shields and conservative design along with airflow passage surveillance for the concrete shield. The shielding properties of all three shields are well known and therefore not subject to significant design uncertainties.

Shielding design is dictated by the regulatory dose limits, maximum bounding radiological neutron and gamma ray source terms of the fuel to be stored, cost, weight (in some cases), and thermal limits of some shielding materials. In general, the commercial nuclear power spent nuclear fuel storage technologies discussed in this appendix were designed to provide adequate shielding for fuel assemblies containing several hundred thousand curies (Ci) of fission products per assembly. The foreign research reactor spent nuclear fuel being considered for acceptance in the United States will contain fission product inventories of from 1,000 to 100,000 (maximum) Ci per assembly. Therefore, the radiation source term for shielding design purposes, assuming the same number of fuel assemblies in each storage technology unit, may be significantly smaller for foreign research reactor spent nuclear fuel than for commercial fuel. The cost savings associated with a reduction in shielding thickness are expected to be more significant for the metal cask and concrete building designs because of their relatively higher costs. At the present time, it appears appropriate to use available designs from the vendors.

Based on the aforementioned vulnerabilities, the best shield would be the metal cask. Concrete shields are judged second best after metal based on their lack of dependence on any active systems. The water shield requires active systems for decay heat removal to prevent heatup and makeup to compensate for long term evaporation. It is also vulnerable to leaks from connected piping and its enclosing structure. Although water appears to be the least expensive shield material, its requirements for several active systems and qualified walls and floor actually make it one of the more expensive shields.

F.2.4 Structural Integrity

All of the spent nuclear fuel storage technologies are required to meet the same standards for structural integrity in accordance with appropriate codes. Structural integrity ensures that the confinement boundary around the spent nuclear fuel is maintained under all operational and accident conditions.

For incident-free operation, the dry storage designs are analyzed in terms of peak stresses on their canister and enclosing structure (i.e., metal cask, concrete cask, or vault). In wet storage designs, the fuel racks and pool structure are analyzed for operating loads. The source of these loads, in accordance with appropriate American Society of Mechanical Engineers codes, include such factors as deadweight, pressure, fill gas pressure, and thermal gradients.

For accident cases, additional loads are imposed upon the structures. These additional loads include seismic acceleration, high (or low) ambient temperature and solar heat flux, component drop or tip over, airflow passage blockage, external fire, tornado missile, flooding, etc. As with incident-free operation, specific prescribed margins of safety between the peak calculated stresses and the maximum allowable stress for a given component, location, and material must be maintained to substantiate structural integrity.

The principal structural-related differences between foreign research reactor spent nuclear fuel and commercial fuel for storage technology design purposes are:

- a typical foreign research reactor spent nuclear fuel element is much lighter [5 kg (11 lbs) as compared to 800 kg (1,760 lbs)] and shorter than a commercial fuel assembly (a stack of 5 typical foreign research reactor spent nuclear fuel elements is approximately equal in length to 1 commercial fuel assembly), and

- the strength of foreign research reactor spent nuclear fuel, in particular the predominant aluminum-clad design, is expected to be less than the commercial fuel assembly.

The much lower foreign research reactor spent nuclear fuel weight will reduce the total weight and load on the storage technology unit by about 19 metric tons (21 tons) for a 24-commercial fuel assembly design. For metal and concrete casks, this is a significant fraction of the total cask's weight, and can only improve the structural strength of the cask. The lower weight of the foreign research reactor spent nuclear fuel will increase the structural margins in the design and possibly allow for the use of less material in the structure compared with the commercial cask design. Any design changes to take advantage of the lower fuel weight would require detailed re-analysis, and are probably unnecessary.

The lower strength of the foreign research reactor spent nuclear fuel would require analyses to demonstrate that operational and postulated accident events do not result in structural failure of the fuel. However, since the principal means of confinement is the canister surrounding the fuel, its structural integrity is expected to be maintained, as it has already been qualified for the heavier commercial fuel under the same conditions and accidents.

Assuming that the same structural design limits apply for foreign research reactor spent nuclear fuel storage as for commercial fuel storage, the lower weight and strength of the foreign research reactor spent nuclear fuel would be expected to increase the original stress design margins.

The basket of any currently licensed cask would require redesign to accommodate the foreign research reactor spent nuclear fuel. Furthermore, it could be anticipated that permanently installed neutron poisons may be required in the basket to prevent criticality for the highly enriched fuels (initially 90 to 93 percent enrichment).

Each of the spent nuclear fuel storage technology designs that have been licensed by the NRC have undergone rigorous structural analyses and have been shown to meet all applicable standards and codes. Designs which have not yet been licensed would be required to present detailed structural analyses for review and confirmation to ensure structural integrity. No design has specific structural vulnerabilities that make it unsuitable for the storage of foreign research reactor spent nuclear fuel. It should be noted that any changes in existing NRC-approved storage designs that are deemed to impact stresses (i.e., reducing shielding wall thickness) would require extensive re-analysis and technical review for structural integrity. Thus, use of existing designs is favored.

F.2.5 Thermal Performance

Adequate decay heat removal is vital to preventing degradation of the fuel cladding barrier to fission product releases. The wet and dry storage technologies rely on a combination of conduction, convection (natural or forced), and radiation heat transfer mechanisms to ensure fuel cladding temperatures below appropriate long term storage limits.

In wet pool designs, fuel decay heat is transferred to the pool water by conduction and natural convection, which is induced by the axial enthalpy rise of the water as it passes over the active region of the fuel. An active cooling system consisting of redundant pumps, heat exchangers, and piping connected to the pool removes the heat in the bulk pool water. Careful thermal design of the spent nuclear fuel storage racks allows for sufficient natural convection flow over each fuel assembly to prevent any local nucleate boiling on the cladding surface throughout the pool. Therefore, the thermal performance of the pool technology relies on storage rack design for local thermal effects and an active external system for global heat removal. As previously discussed, this design has a long-established history of satisfactory performance. Wet storage can accommodate fuel of any power level.

The metal cask, dry storage design relies on a totally passive system for heat removal. The fuel decay heat, in an encapsulating inert gas atmosphere canister, is transferred to the canister's walls by a combination of radiation and conduction heat transfer. The canister walls, in contact with the metal (or sometimes metal sandwiched with a neutron-absorbing material) cask wall transfers this heat by conduction through the metal wall. At the outside of the metal cask, the heat is removed by conduction and natural convection to the environment. Some designs incorporate cooling metal fins on the exterior of the cask to enhance heat transmission to the air. The four metals used in spent nuclear fuel storage cask designs are ductile cast iron, carbon steel, lead, and stainless steel. In terms of their heat conduction properties, cast iron, lead, and carbon steel are superior to stainless steel because they have a thermal conductivity which is about three times that of stainless steel. The metal cask heat transfer system is not susceptible to thermal limits, since these metals have a higher temperature limit than fuel cladding. The only possible degradation of heat transfer could occur if the fuel canister seal was broken and the inert gas atmosphere lost. The sealing system is designed to withstand all postulated accidents and maintain integrity over the lifetime of the cask, because it constitutes part of the radioisotope confinement boundary.

As with metal casks, concrete casks use a passive heat removal system, but the concrete cask system has one inherent vulnerability. To remove fuel decay heat and stay below both the fuel cladding and concrete temperature limits, concrete casks must include a labyrinth airflow passage design that allows natural convection-driven air to enter the cavity enclosing the canister inside the concrete. The air then exits through higher elevation paths through the concrete to the environment. Concrete thermal conductivity is a factor of 10 to 40 lower than that of the previously discussed cask metals. The need for these airflow passages and their associated inlets and outlets introduces the possibility of an accident in which the inlet and/or outlets could be blocked by debris, snow, or even nests or hives. Therefore, concrete casks require surveillance of their air inlet and outlet flow passages. Typically, measurement of the air temperature rise between the inlet and outlet is also used to validate the thermal design and as an operating specification. The elevation difference between the air inlets and outlets is an important design factor in the effectiveness of natural convection-driven airflow through the cask. Larger elevation differences induce a greater airflow rate, which improves heat removal. The above discussion would not apply to a solid concrete design (such as a SILO), because it does not use internal airflow passages for decay heat removal; it uses only concrete conduction.

In a concrete cask, heat transfer within the canister is identical to that of the metal cask. The canister transfers heat by conduction, natural convection, and radiation heat transfer to the airflow around it and the concrete walls surrounding it. That portion of the heat transferred to the concrete walls is then conducted through the concrete to the outside air. The concrete building technology uses a heat transfer system identical to that of the concrete cask. However, its relatively larger size, translating to a larger flow area for air inlets and outlets, makes it less susceptible to flow passage inlet and/or outlet blockage. Also, the concrete building designs typically incorporate a much larger elevation difference between air inlets and outlets than the concrete cask, which further enhances natural convection flow driven heat transfer.

For commercial nuclear fuel, the long-term storage temperature limits are well known (Levy et al., 1987; Johnson and Gilbert, 1983; Einziger and Cook, 1985; and Kohli et al., 1985), and typically about 350°C (662°F). The shielding material limits usually apply to concrete or concrete-like materials, but may also apply to resins or polyethylene. Shielding material temperature limits are not affected by the use of foreign research reactor spent nuclear fuel instead of commercial nuclear fuel in the storage technology. However, the thermal limits of the TRIGA and MTR foreign research reactor spent nuclear fuel could affect the thermal design.

Currently, there is limited well-documented information available on long-term foreign research reactor spent nuclear fuel storage temperature limits for fuel cladding. However, the aluminum cladding of the MTR-type foreign research reactor spent nuclear fuel has a much lower melting point than commercial nuclear fuel zircaloy cladding [649°C (~1,200°F) for aluminum versus 1,832°C (~3,330°F) for zircaloy]. Thus, the maximum long-term storage temperature limit for aluminum-clad foreign research reactor spent nuclear fuel would be expected to be considerably lower than that for zircaloy-clad commercial nuclear power fuel. Aluminum also undergoes a phase change at around 250°C (482°F), which results in a reduction of its tensile properties. An offsetting physical property of aluminum that may partially compensate for its lower melting point is that aluminum has a thermal conductivity more than 10 times greater than zircaloy. This would tend to reduce the temperature difference across the aluminum cladding as compared to zircaloy cladding. TRIGA foreign research reactor spent nuclear fuel cladding is composed of stainless steel or inconel, which have similar thermal conductivities to zircaloy, but a melting temperature of about 1,371°C (2,500°F). TRIGA fuel storage temperature limits are expected to be greater than for aluminum-clad fuel. The Savannah River Site is conducting a research and development project, initiated in FY 1994, to examine the applicability of aluminum-clad spent nuclear fuel dry storage.

At a minimum, a new thermal analysis would need to be performed for existing designs of spent nuclear fuel storage technologies. This analysis would use the parameters associated with foreign research reactor spent nuclear fuel instead of commercial spent nuclear fuel. The important changes in thermal performance parameters for the foreign research reactor spent nuclear fuel are:

- lower individual fuel assembly decay heat power,
- lower and/or different fuel temperature limits for aluminum-clad and stainless steel-inconel-clad fuels, and
- higher clad and fuel thermal conductivity for aluminum foreign research reactor spent nuclear fuel.

A temperature limit of 175°C (347°F) has been tentatively identified to avoid damage to the cladding of aluminum-clad spent nuclear fuel (Shedrow, 1994a and 1994b; Taylor et al., 1994). The results of this revised thermal analysis could impact the thermal design of the spent nuclear fuel storage technology. If the existing design results in unacceptable fuel and/or shielding temperatures, redesign could reduce the maximum heat load of each module or cask or increase the airflow passage area or height for concrete casks that rely on natural convection heat transfer. The new thermal analysis must take into account design restrictions that are imposed by criticality limits (i.e., the maximum allowable number of foreign research reactor spent nuclear fuel assemblies), and possible changes in shielding thickness due to lower gamma and neutron source terms that would improve the storage technology's thermal performance. Again, existing designs should be used to the greatest extent possible.

Commercial spent nuclear fuel dry storage systems require a minimum cooldown period of 5 years. For aluminum-clad foreign research reactor spent nuclear fuel, the preliminary cladding temperature limit of 175°C (347°F) becomes the determining criteria for dry storage loading above an average spent nuclear fuel element power level of 40 Watts each. Foreign research reactor spent nuclear fuel averages more than 40 Watts per element after a single year's discharge from the reactor and, if immediately placed into dry storage, would result in oversized facilities within several years as the radionuclides decay. Consequently, for the size of a foreign research reactor spent nuclear fuel dry storage facility to be minimized, an average foreign research reactor spent nuclear fuel power level below 40 Watts per element is necessary. On average, a 3-year cooldown period would be required. This results in the element's volume being the constraining criteria, and corresponds to maximum density of spent nuclear fuel (hence, minimum size of the facility) in the dry storage method. Consequently, the storage approach uses a minimum wet storage period of 3 years prior to emplacement into dry storage.

A comparative evaluation of the thermal performance of each fuel storage technology points to the metal cask and the solid concrete SILO as the simplest, effective, and least susceptible to any degradation. However, another design which has many merits is the concrete building. Although concrete buildings require open airflow passages to remove decay heat, size and a large elevation difference are factors which compensate for this weakness and make them good candidates. The concrete cask, with adequate design margins and surveillance is an acceptable thermal system. Finally, the wet pool system is a proven technology, but is dependent on an active system to remove heat. The inground concrete system in

Denmark (RISO National Laboratory) relies on a forced air active system and is characterized similar to the wet system in terms of its heat removal capabilities.

F.2.6 Ease of Use

For spent nuclear fuel storage, ease of use is defined as the lack of complexity involved in the process of loading spent nuclear fuel, and operating and maintaining the storage technology. For all storage designs, the spent nuclear fuel must be removed from the transportation cask to be placed into the storage facility, unless the design is a dual-purpose cask.

The technology that requires the fewest steps and lowest complexity for transferring spent nuclear fuel from the transportation cask to its storage location is wet pool storage. At a pool, the transportation cask is simply immersed under the water, opened underwater, and the fuel moved underwater to its final location in a storage rack in the pool. Pool water provides shielding, heat removal, and viewing of the fuel. The dry storage technologies all require additional intermediate steps, which include the insertion of fuel into a canister that must be subsequently drained of all water and air, seal welded, tested for leakage, and backfilled with inert gas. The canister is then placed into its dry storage structure (i.e., vault, concrete, or metal cask). The vault provides for this entire process within a shielded enclosing building, whereas the casks require transport by some vehicle between the transportation cask fuel transfer location and the cask site. Thus, for spent nuclear fuel transfer and loading, the wet storage design is easiest to use, followed by the dry vault.

After loading, operation of the storage facility is another important factor in determining ease of use. For operation, the individual metal or concrete casks are easiest, since they are designed as totally passive systems requiring only periodic visual inspection from a distance. The vault is slightly more complex than the casks because it includes a number of active systems (i.e., crane, power supplies, fuel handling machine) that may require some operational support. The wet storage is the most complex from an operational viewpoint because it includes a number of vital safety-related systems that must be monitored and controlled (e.g., heat removal system, water purification system, makeup water system, ventilation exhaust system).

Maintenance ease of use is closely related to operational ease of use since designs with more operational complexity require greater maintenance. Thus, the cask systems can be considered easiest, followed by the vault, and the wet technology.

In ranking the relative importance of the three aforementioned factors of ease of use (fuel loading, operation, and maintenance) the fraction of time spent during the life of the facility for fuel loading is expected to be much smaller than for operation and maintenance. Fuel loading will be a sporadic event over a long period of time, whereas operation and maintenance are considered continuous over this same period of time. Thus, operation and maintenance ease of use is given greater importance than fuel loading ease of use. With this ranking, the cask (both metal and concrete) technology is judged to have the greatest ease of use, followed by the vault system. The wet storage technology is judged to have the lowest ease of use principally because of its safety-related active systems.

F.2.7 Cost

Information on the cost of different spent nuclear fuel storage technologies is limited because of its proprietary nature, but several comparative statements apply to the different designs. Operation and maintenance costs are expected to be highest for those technologies that rely on active systems for safety. Thus, the wet pool and inground forced air technologies have higher operations and maintenance costs than dry metal casks, dry concrete casks, and dry concrete buildings. The dry concrete vault/building technology would be expected to have slightly higher operations and maintenance costs than the individual metal or concrete casks, since these buildings use active nonsafety systems such as lighting, cranes, and fuel drying dedicated to the vault facility.

For the construction of a new fuel storage facility for the purpose of storing foreign research reactor spent nuclear fuel on the order of approximately 23,000 assemblies, elements and/or rods, it is assumed that 5 "trimmed" foreign research reactor spent nuclear fuel assemblies would occupy the same approximate space as one commercial nuclear power plant fuel assembly (Boiling Water Reactor-type). "Trimmed" means that the non-essential portions (i.e., ends) of the spent nuclear fuel element have been removed, as detailed in Appendix B. Therefore, storage of the total amount of foreign research reactor spent nuclear fuel under consideration in this EIS would be the equivalent of about 5,000 commercial Boiling Water Reactor spent nuclear fuel assemblies. Since typical concrete, inground, or metal casks can store 52 power fuel assemblies, this foreign research reactor spent nuclear fuel inventory would require around 100 casks. A suitably sized single pool or concrete building could accommodate this inventory of spent nuclear fuel. Spent nuclear fuel storage manufacturers have indicated that metal casks typically cost about twice as much as concrete casks for the same quantity of fuel storage due to the higher costs of metal as compared to concrete. Based on its design, the least expensive concrete cask is expected to be the simple concrete SILO, since it does not have steel-lined internal air passages. The number of foreign research reactor spent nuclear fuel assemblies under consideration in this EIS may be amenable to the economic advantages that a single building or pool offers over a large number of individual casks. Another potential cost advantage of the pool or concrete building is that these are self-contained, not requiring access to any other facilities for the transfer of the fuel from the transport cask. Presented below is a brief summary of commercial cost experience with storage.

F.2.7.1 Costs for Dry Storage Designs

The cost for different spent nuclear fuel storage technologies varies significantly between designs. Some information on cost has been obtained from manufacturers and openly available literature. Relative order of magnitude cost data was obtained for the horizontal concrete NUHOMS module, vertical concrete Ventilated Storage Cask design, vertical concrete SILO, metal CASTOR vertical cask, and the modular dry vault concrete building design.

The principal elements of cost that should be considered for the storage of foreign research reactor spent nuclear fuel are: (1) engineering for redesign and licensing, (2) capital for the construction of the facility, and (3) operations and maintenance. In the interest of minimizing cost and schedule for the completion of any storage facility for foreign research reactor spent nuclear fuel, the licensing basis of 10 CFR 72 used by the NRC for commercial nuclear power plant spent nuclear fuel should be adopted for the foreign research reactor spent nuclear fuel. This regulation provides all the requirements for licensing foreign research reactor spent nuclear fuel storage and has been successfully applied to numerous dry spent nuclear fuel storage installations in the United States.

Redesign engineering should be limited to changes in the design of the basket that encapsulates the fuel, since foreign research reactor spent nuclear fuel has different dimensions, would probably be stacked, and could require different spacing and/or the incorporation of neutron absorbing plates to maintain subcriticality safety margins. Outside the basket, all remaining components should be identical to those already licensed for commercial nuclear fuel by the NRC, thereby significantly reducing engineering analysis and license review time and costs as well as drawing and specification changes. This could result in some overdesign in the shielding and heat removal of the system, but would have the benefit of greatly reduced engineering, licensing, and schedule costs. If thermal analyses show that unacceptable foreign research reactor spent nuclear fuel temperatures would occur in the storage facility, then more extensive redesign would be required (e.g., reduce excess concrete wall thickness not needed for shielding, which then improves the conduction heat transfer) or fewer fuel assemblies could be stored in each unit.

Information obtained on the unit capital costs for different storage designs shows a significant variation. The least expensive unit is the SILO due to its simple concrete-canister design and lack of internal air passage labyrinth. The most expensive unit cost, excluding the modular dry vault (which stores a larger number of fuel assemblies than the other storage designs), is the CASTOR metal cask, due to its use of a thick metal wall instead of concrete. The Ventilated Storage Cask and NUHOMS designs' costs fall between the SILO and CASTOR. If one were to rank, in decreasing order, the unit cost of the four cask designs, they would be: CASTOR, NUHOMS, Ventilated Storage Cask, and SILO. There is more than a factor of 10 difference between the SILO and the CASTOR.

An estimate of the capital costs for storing approximately 23,000 foreign research reactor spent nuclear fuel elements can be made with the following two assumptions. First, the average spent nuclear fuel assembly decay heat is between 10 and 40 Watts, which is reasonable and conservative based on the status of foreign research reactor spent nuclear fuel under consideration in this EIS. Second, five trimmed foreign research reactor spent nuclear fuel assemblies can be stacked to fit into the same approximate space as one commercial nuclear power plant spent nuclear fuel assembly (Boiling Water Reactor-type). Using these assumptions, 23,000 foreign research reactor spent nuclear fuel elements would require 375 SILOs, 100 VSC-24s, 100 NUHOMS-24Ps, or 150 CASTOR V21s. One sufficiently sized and designed modular dry vault would also accommodate the foreign research reactor spent nuclear fuel.

The operations and maintenance costs for all these designs are expected to be small based on utility experience in operating dry spent nuclear fuel storage at numerous sites throughout the United States. The passive nature of these designs eliminates the need for any control room or continuous monitoring. Once completed and loaded with fuel, the storage facility would not require any onsite staff. Remote security surveillance cameras, fences, and thermoluminescent dosimeters for radiation monitoring would be utilized. Some designs (Ventilated Storage Cask and NUHOMS) require a periodic visual inspection of the labyrinth airflow inlets and outlets to ensure that there is no blockage. Periodic security fence thermoluminescent dosimeter retrieval and analysis would also be an expected operating requirement. No onsite utility consumption would be necessary except for that used in security lighting and cameras. Under incident-free conditions, no significant maintenance costs would be anticipated for most of the designs, with the exception of the modular dry vault, where equipment used in the movement and encapsulation of fuel would require some periodic maintenance.

Table F-15 provides a summary of the dry storage costs. Full-Time Equivalent estimates assume full-time assignment, whereas the utility experience indicates only part-time assignment would be necessary. Thus, Table F-15 costs are extremely conservative.

F.2.7.2 Costs for Wet Storage Designs

Based on previous utility experience, the costs associated with the design, licensing, construction, operation, and maintenance of a new spent nuclear fuel pool are expected to be higher than most of the dry storage designs. This is due to the structural, equipment, and active system requirements of a pool that must also be enclosed in a properly qualified structure. The need for active operating safety-related systems at a wet facility increases the operations and maintenance costs. These important systems include water purification and chemistry, water heat removal, water level, and heating, ventilation, and air conditioning for the building. Table F-16 provides a summary of the wet storage costs. Thus, it can be stated that the overall costs associated with the selection of a new wet storage facility for foreign research

Table F-15 Summary of Dry Storage Facility Costs Based Upon Utility Experience[a]

	Approximate Unit Capital Cost Range, $	*Approximate # of Canisters/Sleeves for Foreign Research Reactor Spent Nuclear Fuel*	*Total Capital Cost $M*	*Full-Time Equivalents, Loading/ Inspection[b]*	*Full-Time Equivalents Monitoring[c]*	*Other Annual Costs, $M*	*Total Annual Operating Cost $M[d,e]*
Metal Cask	800,000-1.1M	150 (max)	165	15 (max)	3 (max)	1	3.7
Horizontal Dry Storage Cask	400,000-500,000	100 (max)	50	15 (max)	3 (max)	1	3.7
Vertical Concrete Storage Cask	350,000	100 (max)	35	15 (max)	3 (max)	1	3.7
Modular Dry Vault	13,000/tube	5 foreign research reactor/vaul t tube	65	15 (max)	3 (max)	1	3.7
SILO	100,000	375 (max)	37.5	15 (max)	3 (max)	1	3.7

Reference for costs: (EPRI, 1993)

[a] *Intermediate wet pool required for dry storage facility not included because utilities already possess an on-site pool*

[b] *One shift operation*

[c] *Monitoring based on one Full-Time Equivalent per shift*

[d] *Average Full-Time Equivalent cost of $150,000 per year*

[e] *Estimated absolute maximum from utility experience; around $1 million per year appears to be the average*

Table F-16 Summary of Wet Storage Costs

Approximate Facility Capital Cost	$80-100 million
Full-Time Equivalents Loading/Inspection	30
Full-Time Equivalents Operations/Monitoring	(in above)
Other Annual Costs	around $1million
Total Annual Operating Cost	$6-12 million

(Nuclear Fuel, 1994a)

reactor spent nuclear fuel could be significantly larger than for any comparably sized dry storage design using concrete.

F.2.8 Design, Construction, and Operational Requirements

The DOE has orders dealing indirectly with the storage of spent nuclear fuel. A search was also made to determine if other Federal requirements exist that specifically deal with spent nuclear fuel.

DOE Order 5400.1 (DOE, 1988), entitled "General Environmental Protection Program," establishes environmental protection program requirements. It applies to all Departmental elements and contractors performing work for DOE. Although the order makes no direct reference to spent nuclear fuel, Chapter IV deals with environmental monitoring requirements, which could be useful in establishing a legal framework for storing spent nuclear fuel.

DOE Order 5400.5 (DOE, 1990), entitled "Radiation Protection of the Public and the Environment," establishes standards and requirements for operations of the DOE and DOE contractors with respect to protection of the public from undue radiological risk. The provisions apply to all Departmental elements. This order references the storage of spent nuclear fuel. It also references instances where some DOE facilities are subject to provisions of 10 CFR 72. It was not made clear in the order which DOE facilities are subject to 10 CFR 72, which deals directly with all aspects of interim storage of spent nuclear fuel.

DOE Order 5480.22 (DOE, 1992b), entitled "Technical Safety Requirements," requires that DOE nuclear facilities delineate criteria, content, scope, documents, etc. The scope includes DOE elements, but excludes facilities exempt from NRC licensing and Naval Propulsion Program facilities. Although this order does not reference spent nuclear fuel, there are useful discussions of limiting conditions for operation of nonreactor nuclear facilities.

DOE Order 5633.3A (DOE, 1994e), entitled "Control and Accountability of Nuclear Materials," prescribes minimum requirements and procedures for control and accountability of nuclear materials at DOE facilities, which are exempt from NRC licensing requirements. By DOE definition, "nuclear materials" includes spent nuclear fuel. Storage of nuclear material is mentioned with respect to repositories.

DOE Order 6430.1A, (DOE, 1989a), entitled "General Design Criteria," has a section dealing with irradiated fissile material storage facilities (Section 1320). General criteria for nuclear criticality, confinement systems, effluent control and monitoring, and decontamination and decommissioning are discussed. Reference is made that, "the design professional shall consider the criteria provided in 10 CFR 72," (NRC, 1994) as well as NRC Regulatory Guides 3.49 (NRC, 1981) and 3.54 (NRC, 1984) for applicability to irradiated fissile material storage facilities. Other important standards for dry storage are ANSI/ANS-57.9 (ANSI, 1984a) and NRC Regulatory Guide 1.13 (NRC, 1975).

F.2.9 Aluminum-Clad Research Reactor Spent Nuclear Fuel Dry Storage Experience

F.2.9.1 Australia

Australia has successfully operated an underground dry storage facility for High Flux Australian Reactor MTR-type aluminum-clad research reactor spent nuclear fuel for 31 years at the Lucas Heights Facility (Australia, 1993; Ridal, 1994; Silver, 1993). The facility consists of a building enclosing a concrete floor with 50 steel plugs that are bolted to a steel collar set into the concrete.

Each plug covers a stainless steel-lined 0.64 cm (0.25 in) thick, 14 cm (5.5 in) inner diameter, 15.2 m (50 ft) deep borehole tube that is sealed at the bottom. The rock around these 50 borehole tubes is sandstone with a variable clay matrix and bands of enriched siderite. The actual boreholes in the sandstone are 16.5 cm (6.5 in) in diameter, 16.8 m (55 ft) deep, and spaced 1.14 m (45 in) center-to-center apart.

Each borehole liner is filled with 11 stainless steel canisters that hold 2 stacked fuel assemblies each. The borehole liner is evacuated and backfilled with dry nitrogen. The borehole liner plug is designed with its own plug to allow for atmosphere purging, backfilling, and annual monitoring of any fission product gases that would indicate canister breach.

The stored High Flux Australian Reactor spent nuclear fuel is uranium-aluminum alloy with aluminum cladding in the shape of four concentric tubes. Each fuel assembly has an outer diameter of 10 cm (3.93 in) and a length of 66 cm (26 in). The ^{235}U content for each fresh fuel assembly was 170 g (0.37 lbs), and the ^{235}U was enriched to 60 percent. Fuel has been stored at this facility for 8 to 31 years with no radioactivity releases or evidence of corrosion over this time period. No nuclear poisons for criticality safety or heat transfer analyses were deemed necessary because of the relatively low ^{235}U content, large borehole spacing, and low fuel assembly decay heat. The storage criteria for each fuel assembly is a maximum decay heat of less than 4.5 Watts, which after 20 years, drops to 1.5 Watts per fuel assembly. Fuel examined in a hot cell after 10 and 25 years of storage at this facility showed no visible signs of corrosion. Figure F-29 illustrates the facility design.

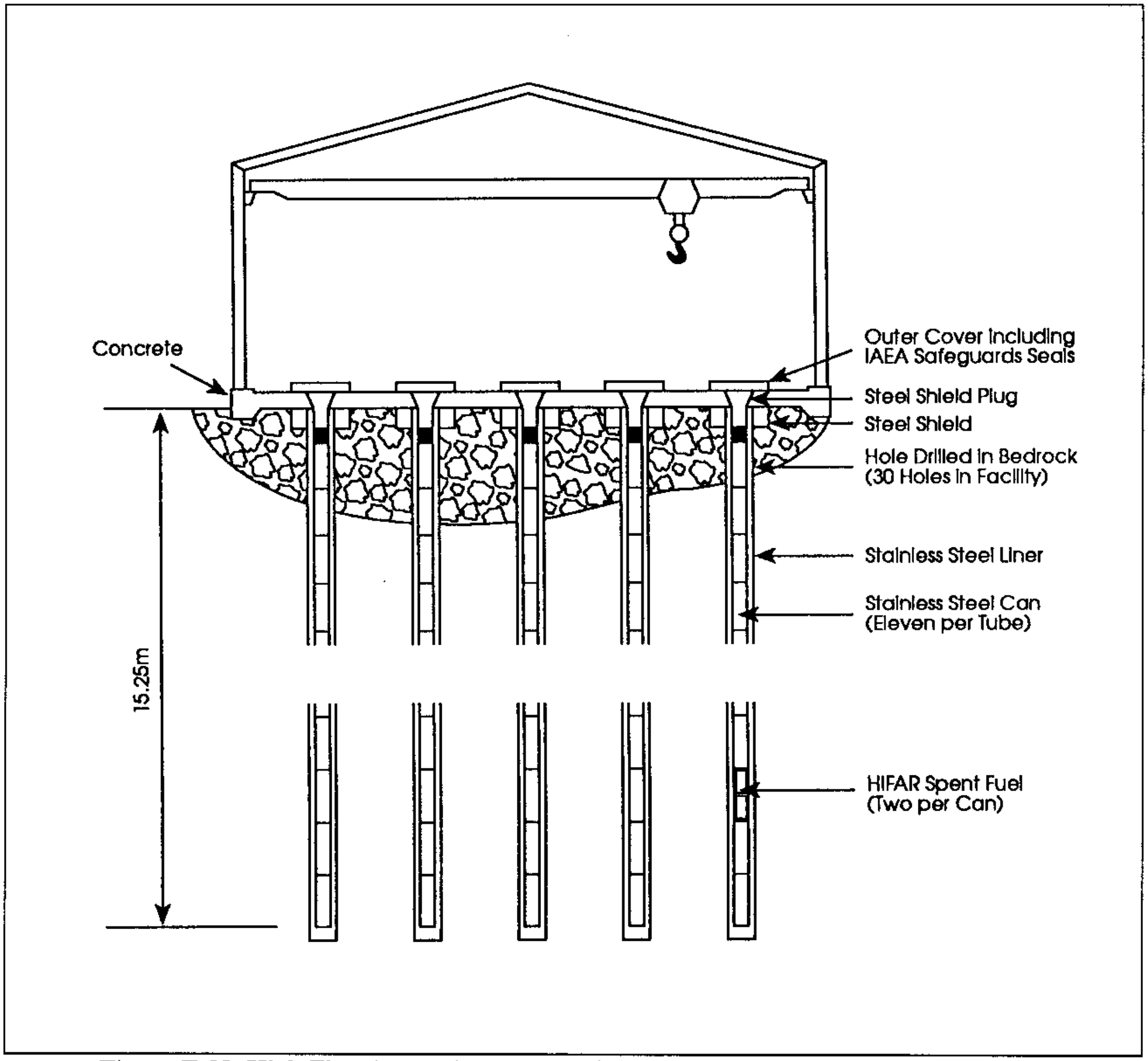


Figure F-29 High Flux Australian Reactor Spent Nuclear Fuel Dry Storage Facility

F.2.9.2 Japan

In 1982, The Japan Atomic Energy Research Institute completed construction of a dry spent nuclear fuel storage facility at Tokai, Japan for the storage of JRR-3 research reactor spent nuclear fuel (Shirai et al., 1991). The facility consists of a building enclosing several support areas (cask receipt, loading, cask maintenance, and control room) and the drywell storage structure (Figure F-30). The storage structure is 12 m (39.4 ft) long, 13 m (42.7 ft) wide, 5 m (16.4 ft) high concrete box that encapsulates a 10 x 10 lattice array of drywells (Figure F-31). Each drywell storage canister (Figure F-32) comprises a

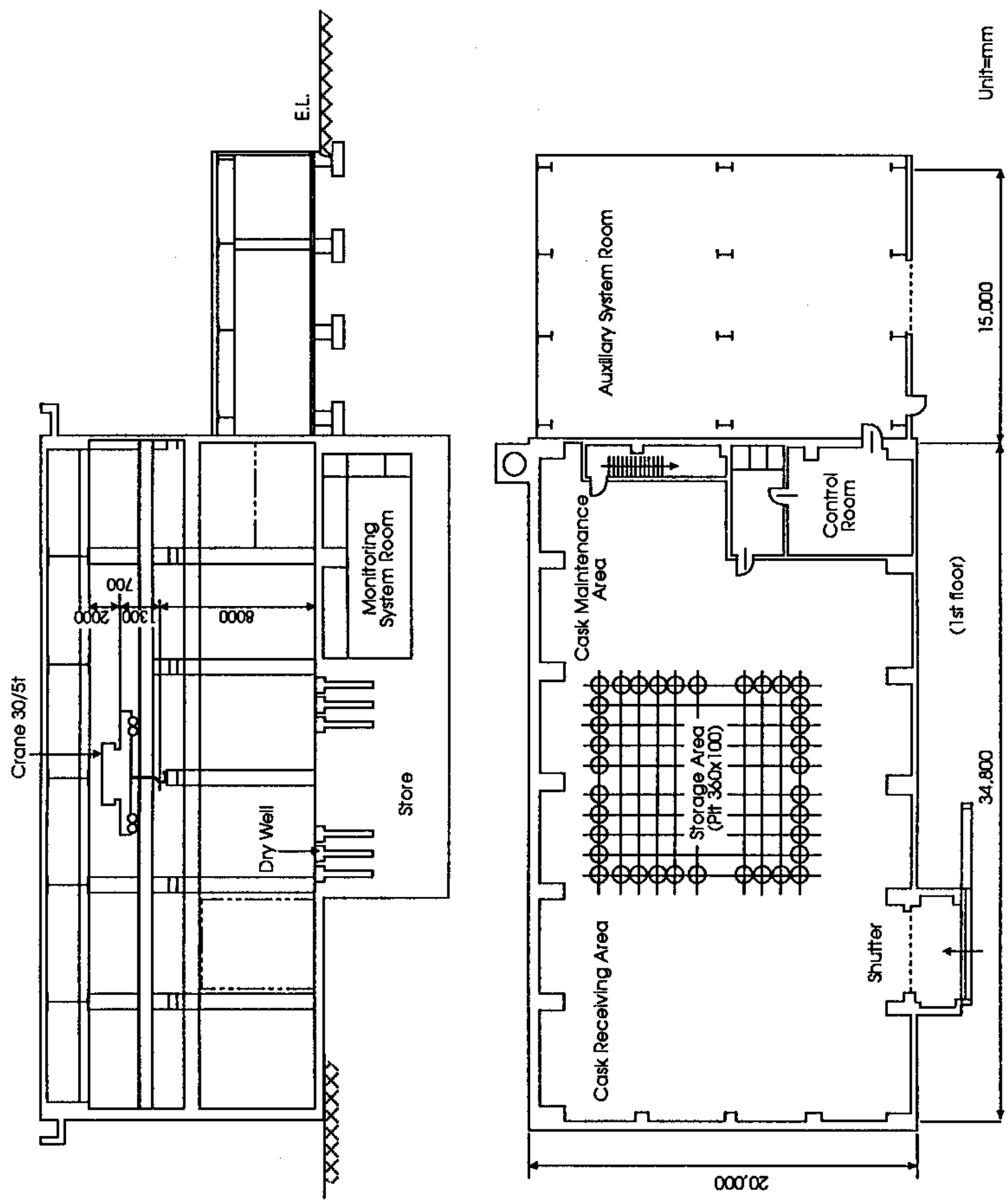


Figure F-30 General Arrangement of Dry Storage Facility at Tokai, Japan

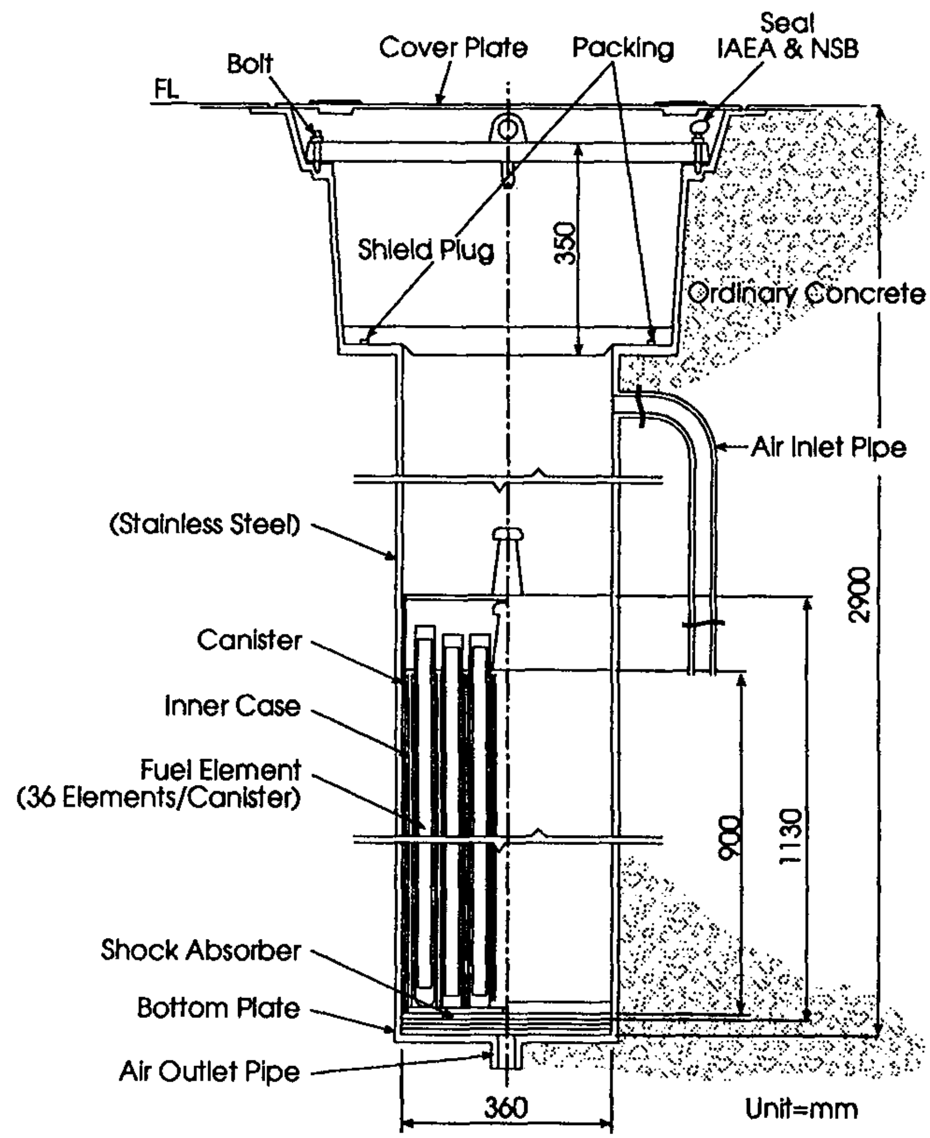


Figure F-31 JRR-3 Dry Storage Facility

0.8 cm (0.3 in) thick stainless steel liner 2.5 m (8.2 ft) deep and has a 36 cm (14.2 in) inner diameter. Each drywell has a labyrinth air inlet and outlet pipe for radiation monitoring and decay heat removal, and is covered with a 35 cm (13.8 in) thick carbon steel shield plug. The plug is bolted to the concrete and has a cover plate above it. Each drywell has a minimum of 1.5 m (4.9 ft) of concrete shielding around it.

A cylindrical stainless steel canister (Figure F-32) is placed in each drywell. The canister has 0.5 cm (0.2 in) thick walls, a 35 cm (13.8 in) outer diameter, and a height of 1.25 m (4.1 ft). Each canister holds 36 fuel elements and is fusion welded after being loaded with spent nuclear fuel, evacuated, and filled with inert gas. Each element is a natural metallic or 1.5 percent ^{235}U-enriched uranium oxide cylinder encased

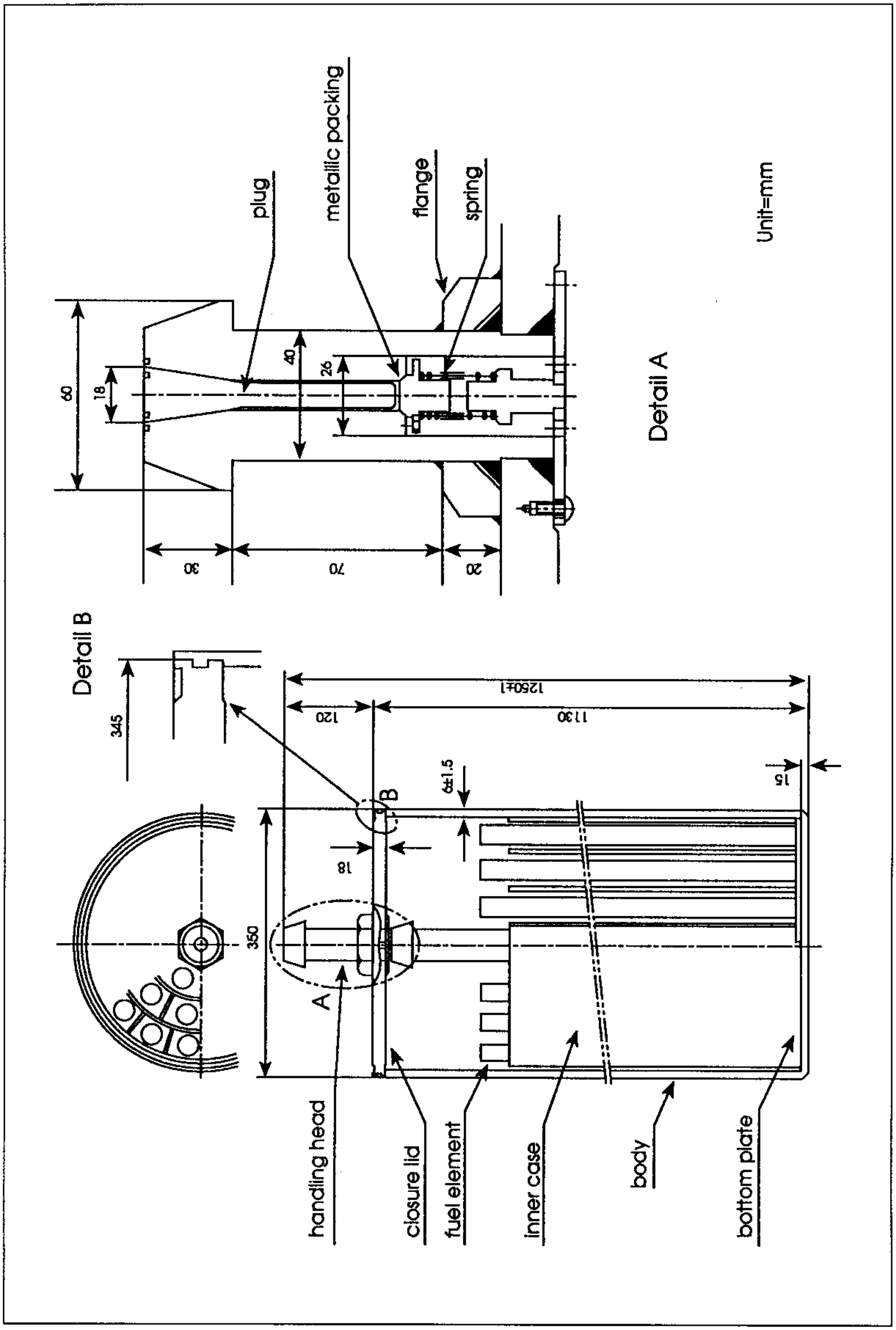


Figure F-32 Storage Canister

in aluminum cladding. The element is about 95 cm (37.4 in) long, with a 2.5 cm (1 in) outer diameter. Before storage, each element must meet the specifications of:

- maximum burnup = 800 Megawatt Days per metric ton,
- minimum cooling time = 2,500 days (6.8 years),
- maximum fission product activity = 110 Ci, and
- maximum decay heat = 0.5 Watt.

Although the low decay heat for the fuel elements eliminates the need for any cooling system (i.e., natural convection and conduction are sufficient), a system of blowers, filters, dehumidifiers, and monitors is provided for the facility. This system is used to provide subatmospheric pressure in the drywell so that any leakage would always be into the structure and its associated filters. This system ensures low humidity for minimizing any corrosion, and is also part of the radiation monitoring system designed to sample airflow for any escaping Krypton-85, a long-lived fission product that would be indicative of canister and fuel degradation. Maximum fuel storage temperature is maintained below 45°C (113°F) in this storage facility.

As of 1991, 1,800 fuel elements [14 metric tons (15 tons) of uranium] have been stored at this facility without any incidents. In 1987, after 5 years of storage, 2 canisters and their 72 fuel elements were removed and examined. None of the fuel elements or canisters exhibited any signs of corrosion, cracking, degradation, or failure.

F.2.10 Summary

In summary, the preceeding discussion indicates the following:

- Dry storage of spent nuclear fuel is a mature technology and requires the least maintenance.
- Dry storage of foreign research reactor spent nuclear fuel appears to be practical using existing designs from commercial utility experience and has been demonstrated in operating storage facilities for foreign research reactor spent nuclear fuel in Australia and Japan.
- Wet storage technology is the most common method for spent nuclear fuel.

F.3 Selection of Storage Technologies for Further Evaluation

The preceding discussions have identified several storage technologies suitable for foreign research reactor spent nuclear fuel. Three basic categories encompass these technologies:

- Dry Vault Storage,
- Dry Cask Storage, and
- Wet Pool Storage.

The three technologies are discussed in this section, including site-specific modifications, while Section F.4 describes the potential impacts and ramifications at the five candidate management sites. All three approaches are estimated to require less than 4.5 ha (11 acres) of site land for receipt of all of the foreign research reactor spent nuclear fuel under consideration in this EIS.

F.3.1 Dry Storage Facility Designs

F.3.1.1 Spent Nuclear Fuel Storage Using a Dry Vault (Modular Dry Vault Storage)

As noted previously, the dry vault facility is an aboveground, self-contained concrete structure that includes dry fuel loading and unloading (Fort St. Vrain, 1992; Shedrow, 1994a and 1994b; Taylor et al., 1994; Claxton et al., 1993). The vault approach design consists of four major components: a receiving/loading area, fuel storage canisters, a shielded container handling machine, and a modular array for storing the fuel storage canisters (Figure F-33). The receiving area uses a small wet pool for unloading the transportation casks and for short-term (1 to 3 year) storage of foreign research reactor spent nuclear fuel exceeding 40 Watts per element. Table F-17 summarizes some typical modular dry vault storage parameters. The vault consists of several array units, and each unit provides storage for hundreds of fuel elements. The vault itself consists of a charge/discharge bay with a fuel handling machine above a floor containing steel tubes that house the (removable) fuel canisters. Shielding above the spent nuclear fuel is provided by the thick concrete floor and shield plugs inserted into the top of the steel storage tubes. The steel tubes serve as secondary containment for the foreign research reactor spent nuclear fuel and descend into an open storage area. Large labyrinth air supply ducts and discharge chimneys permit natural convection cooling of the steel spent nuclear fuel storage tubes, while the perimeter concrete walls provide for shielding. The design allows for expansion by adding additional units of arrays to the end of the vault, or by construction of another module. The vault facility also includes a receiving and loading bay that allows handling of the shielded transportation casks and unloading of the foreign research reactor spent nuclear fuel into the short-term wet storage pool. The receiving bay provides for spent nuclear fuel inspection, canning as required, and could be used for spent nuclear fuel characterization with additional equipment and modifications.

In operation, the transportation cask is lifted by a crane and placed in the unloading area of the small wet pool. The fuel elements are removed under water, examined, and, if the heat generation rate is below 40 Watts per element, the spent nuclear fuel is placed within the transfer canister. The transfer canister is subsequently drained, dried, and seal-welded. The handling machine then transports the loaded canister to the storage tubes. The handling machine includes radiation shielding. Heat dissipation is accomplished by natural convection from the surfaces of the handling machine and canister. The handling machine transfers the spent nuclear fuel canister from the receiving area to the vault and places the canister vertically into the storage tubes. The shield plug is placed on top of the loaded storage tube. Decay heat is dissipated by natural convection; air enters through inlet ducts at the bottom of the vault module, passes around the outside of the steel storage tubes containing the spent nuclear fuel canisters, and exits through outlet ducts at the top of the module. The vault facility stores spent nuclear fuel in canisters that are approximately 40.6 cm (16 in) in diameter by some 4.6 m (15 ft) long. As currently envisioned, foreign research reactor spent nuclear fuel would be stored in 5 levels of fuel with 4 elements per level, for a total of 20 fuel elements per spent nuclear fuel canister (MTR aluminum-clad type design). The vault design allows for 36 to 44 canisters per array unit, depending upon the decay heat of the spent nuclear fuel and a cladding temperature limit [175°C (347°F) for aluminum cladding with an air inlet temperature of 49°C (120°F)] (Shedrow, 1994a; Taylor et al., 1994).

Thus, the number of vault units/arrays required is as follows:

- spent nuclear fuel decay heat exceeding 80 Watts per element: 35 vault units;

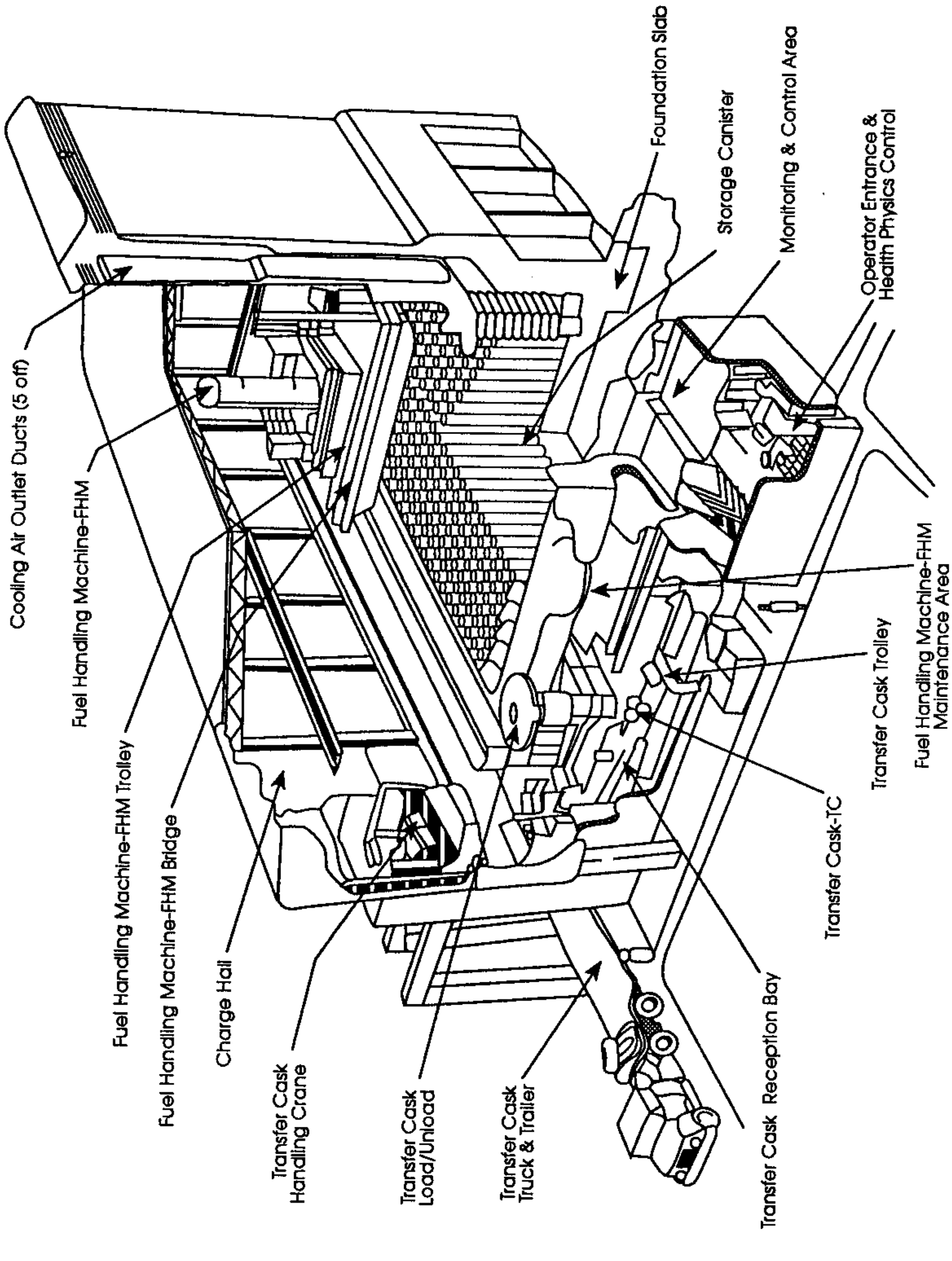


Figure F-33 Vault Elevation View

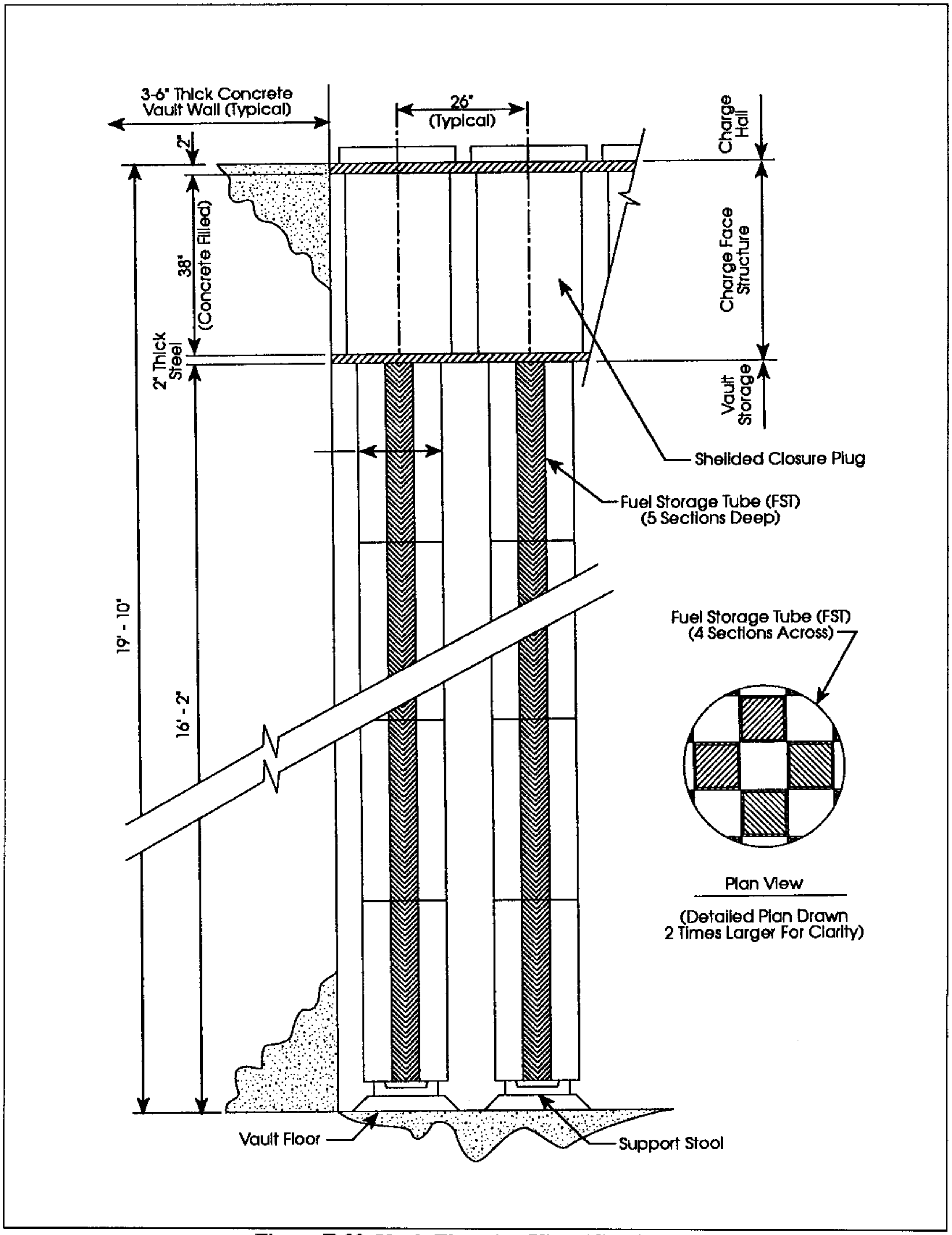


Figure F-33 Vault Elevation View (Continued)

Table F-17 Summary of Modular Dry Vault Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel[a]

Construction Phase:	
Disturbed Land Area	3.7 ha (9 acres)
Facility: Size (Area) Concrete Steel	 5,000 m^2 (54,000 ft^2) 21,800 m^3 (28,500 yd^3) 5,200 mt (5,750 tons)
Soil Moved	11,000 m^3 (14,400 yd^3)
Equipment Fuel	835,000 l (221,000 gal)
Construction Debris/Waste	1,800 m^3 (2,400 yd^3)
Work Force	190/yr average, 234/yr peak
Duration (Years)	4 years for construction, 1.5 years for design
Capital Cost	$370 million[b]
Operation Phase:	
Electricity	800 - 1,000 MW-hr/yr (staging facility)
Water	2.1 million l/yr (550,000 gal/year) for first 10 years, 0.9 million l/yr (238,000 gal/yr) thereafter
Wastestreams	
Solid Low Level Waste	22 m^3/yr (780 ft^3/yr) during receipt, 1 m^3/yr (35 ft^3/yr) thereafter
Waste Water	1.59 million l/yr (420,000 gal/yr) during receipt, 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30 during receipt, 8 thereafter
Annual Operating Cost	$15.6 million during handling, $0.6 million during storage[b]

[a] *Staging facility parameters are based upon the regionalized, small wet pool (Dahlke et al., 1994).*

[b] *Cost estimates are in $1993 (EG&G, 1993)*

- spent nuclear fuel decay heat between 40 and 80 Watts per element: 32 vault units; and
- spent nuclear fuel decay heat between 10 and 40 Watts per element: 28 vault units.

For "cold" fuel (10 Watts per element), potentially more than 44 spent nuclear fuel canisters could be placed per vault unit. This would require a customized design. Figure F-34 displays the 10 to 40 Watts and 80 Watts per element cases.

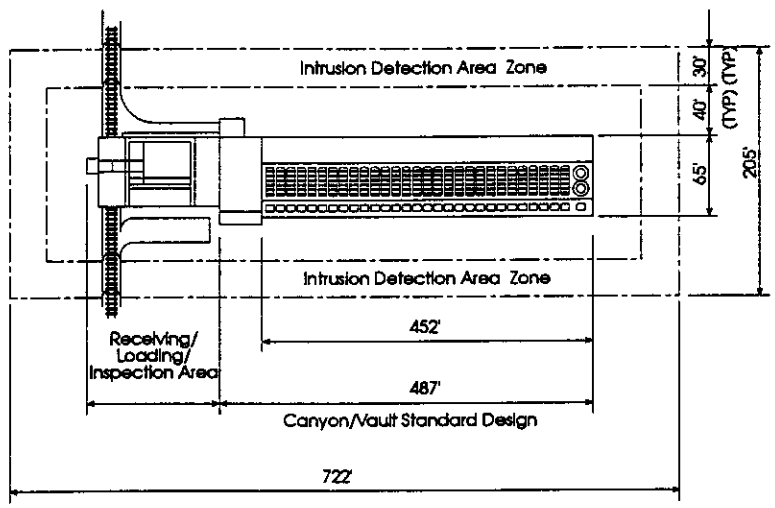
Higher decay heat foreign research reactor spent nuclear fuel would have to be temporarily stored in the small wet storage pool. The storage period is not expected to exceed 3 years.

Criticality concerns are addressed by fuel geometry within the canister and by the use of nuclear poisons (e.g., borated steel in the baskets, etc.). Vault geometry is used to maintain a minimum spacing between adjacent fuel elements or groups of fuel elements to prevent criticality. Nuclear poisons absorb neutrons, thus preventing criticality.

The vault/canyon design has been licensed by the NRC for a specific site. It represents a complete stand-alone facility that can be dedicated to foreign research reactor spent nuclear fuel without requiring the utilization of any other facilities at the host site. Cask handling, maintenance, spent nuclear fuel loading, spent nuclear fuel inspection, spent nuclear fuel storage, and (potentially) characterization can all be accomplished within the same facility.

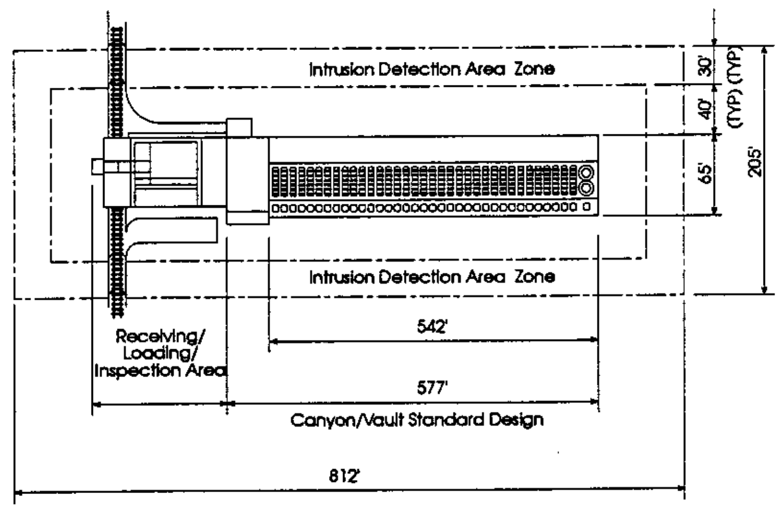
The cost to construct a modular dry vault storage facility with a staging area sufficient to unload, characterize, can, temporarily store in a small pool, and transfer the spent nuclear fuel to the vault storage

Note:
All dimensions shown are approximate and may only be used for a conceptual "footprint" estimate.



10-40 W/Element Scenario
25,000 FRR fuel element canyon/vault (Nominal) Storage req. (29 vaults with 44 tubes per module - 20 FE per tube)
• Passive cooling • Monitored storage canisters • Retrievable for periodic inspection • Self-contained facility for staging and storage of spent fuel elements

Note:
All dimensions shown are approximate and may only be used for a conceptual "footprint" estimate.



80 W/Element Scenario
25,000 FRR fuel element canyon/vault (Nominal) Storage req. (35 vaults with 44 tubes per module - 20 FE per tube)
• Passive cooling • Monitored storage canisters • Retrievable for periodic inspection • Self-contained facility for staging and storage of spent fuel elements

Figure F-34 Canyon/Vault Standard Design (10 to 40 Watt and 80 Watt Scenarios)

area is estimated to be $370 million. The annual operating cost for this facility is estimated to be $15.6 million during the period of handling and transfers of the spent nuclear fuel and $0.6 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993) with the addition of the cost of a small wet storage facility reported by Dahlke et al. (Dahlke et al., 1994).

F.3.1.2 Spent Nuclear Fuel Storage Using Dry Casks

The dry cask storage approach consists of the following components (BG&E, 1989; Duke Power Company, 1988; Shedrow, 1994a and 1994b; Taylor et al., 1994; Claxton et al., 1993):

- a staging facility for cask receipt and unloading and for loading foreign research reactor spent nuclear fuel into the dry storage casks [a wet pool is used for this purpose, and for short-term (1 to 3 years) storage of foreign research reactor spent nuclear fuel with a heat load exceeding 40 Watts per element],
- an inspection/characterization facility, for examining fuel integrity and canning degraded spent nuclear fuel as required (this may be incorporated into the staging facility as an inspection cell or be immediately adjacent to it),
- a dry storage cask (usually concrete) [this provides for the shielding and the structural stability of the spent nuclear fuel storage],
- a transfer mechanism, such as a dedicated truck/trailer combination with a ram for horizontal modules, or a crane for vertical modules, and
- a separate fuel canister which may or may not be used [if used, it is typically around 4.6 m (15 ft) long and 1.7 m (5.5 ft) in diameter and weighs around 32 metric tons (36 tons)].

The dry cask approach requires the staging facility to receive and inspect the spent nuclear fuel shipment. The transportation cask would be unloaded in a small wet pool within the facility. Subsequently, the spent nuclear fuel is loaded into the dry cask (or spent nuclear fuel canister for the horizontal cask), and the cask is placed upon an outside concrete slab. The horizontal approach uses a dry spent nuclear fuel transfer canister for containing the spent nuclear fuel. This is placed within a shielded transfer cask and moved to the outside modular storage facility. A hydraulic ram inserts the transfer canister inside the horizontal storage module, followed by sealing with a shield plug.

The dry storage modules are designed to withstand normal loads and design basis accident effects, such as earthquakes, tornadoes, and floods. The concrete provides radiation shielding for gamma rays and neutrons. Natural air circulation dissipates the heat as air enters through inlet vents near the bottom of the cask, passes around the spent nuclear fuel canister, and exits near the top. Screens and grills keep birds and other animals out of the cooling duct area. Some of the candidate sites have facilities which may be used for cask receipt and unloading and spent nuclear fuel inspection and transfer to storage.

The application of dry cask storage technology to foreign research reactor spent nuclear fuel depends upon the heat load. Horizontal casks are anticipated to be slightly more restrictive than the vertical casks with respect to the heat load, and are thus the focus of the discussion. The standard design for a horizontal fuel canister provides for 24 or 52 sleeves (i.e., Pressurized Water Reactor or Boiling Water Reactor spent nuclear fuel, respectively), each about 4.6 m (15 ft) long. As with the vault approach, it is conservatively assumed that each sleeve contains five foreign research reactor spent nuclear fuel elements (i.e., in layers), within a basket or can arrangement for maintaining spacings and retrievability. As with the vault approach, the number of dry storage casks depends upon the decay heat of the spent nuclear fuel and a cladding temperature limit [175°C (347°F) for aluminum cladding, with an air inlet temperature of 49°C (120°F)]. The 24-sleeve design allows for a maximum of 120 elements of foreign research reactor spent nuclear fuel with 40 to 80 Watts per element of decay heat, while the 52-sleeve design provides for a maximum of 260 elements per dry storage cask. Thus, the number of casks required is as follows:

- decay heat between 40 and 80 Watts per element: 205 casks, and
- decay heat between 10 and 40 Watts per element: 94 casks.

Note that these values are very conservative and correspond to a maximum of around 40 percent of the NRC-licensed heat loads per cask. Initially, foreign research reactor spent nuclear fuel with higher heat loads could be unsuitable for the dry storage cask pending detailed heat transfer analysis and a determination of limiting fuel storage temperature for aluminum-clad and TRIGA spent nuclear fuel. However, such relatively high decay heat fuel represents a small percentage of the currently identified foreign research reactor spent nuclear fuel so that its impact would be small; and after 1 to 5 years of wet storage, it would all be below a heat duty of 80 Watts per elements. The storage approach assures a minimum spent nuclear fuel wet storage time of 3 years after discharge prior to dry storage. This would essentially ensure that all foreign research reactor spent nuclear fuel is below a heat output of 40 Watts per assembly.

Figure F-35 displays approximate layouts for the dry cask storage facility predicated upon a horizontal cask design. Table F-18 summarizes some general parameters of dry cask storage.

The dry storage cask technology requires a separate staging facility for foreign research reactor spent nuclear fuel unloading, canning, and storage cask loading and transportation cask maintenance. This facility has the following operational areas:

- *Transportation Cask Handling*: This incorporates cask maintenance, truck/railcar unloading, decontamination/washdown, radioactive material control, and cask sampling/flushing/degassing.
- A small wet pool for fuel transfer and short-term storage.
- *Spent Nuclear Fuel Unit Handling*: Fuel removal, decontamination, fuel drying, fuel canning, inerting, and thermal measurements.
- *Spent Nuclear Fuel Unit Transfer*: This constitutes placement of the spent nuclear fuel into the cask or canister, followed by sealing.
- *Radwaste Treatment*: This includes collection, treatment, and preparation for disposal of contaminated effluents and radwaste treatment and solidification.
- *Heating, Ventilation, and Air Conditioning*: This represents heating, ventilation, and air conditioning of the facility so that contamination of the workers and the environment is avoided.

The inspection/characterization facility includes a shielded dry hot cell for spent nuclear fuel analysis and examination, and canning of degraded spent nuclear fuel. All equipment and instrumentation within the cells is remotely operated to provide chemical, physical, and radiological properties, as needed. The facility is maintained under negative pressure with exhaust through High Efficiency Particulate Air filters

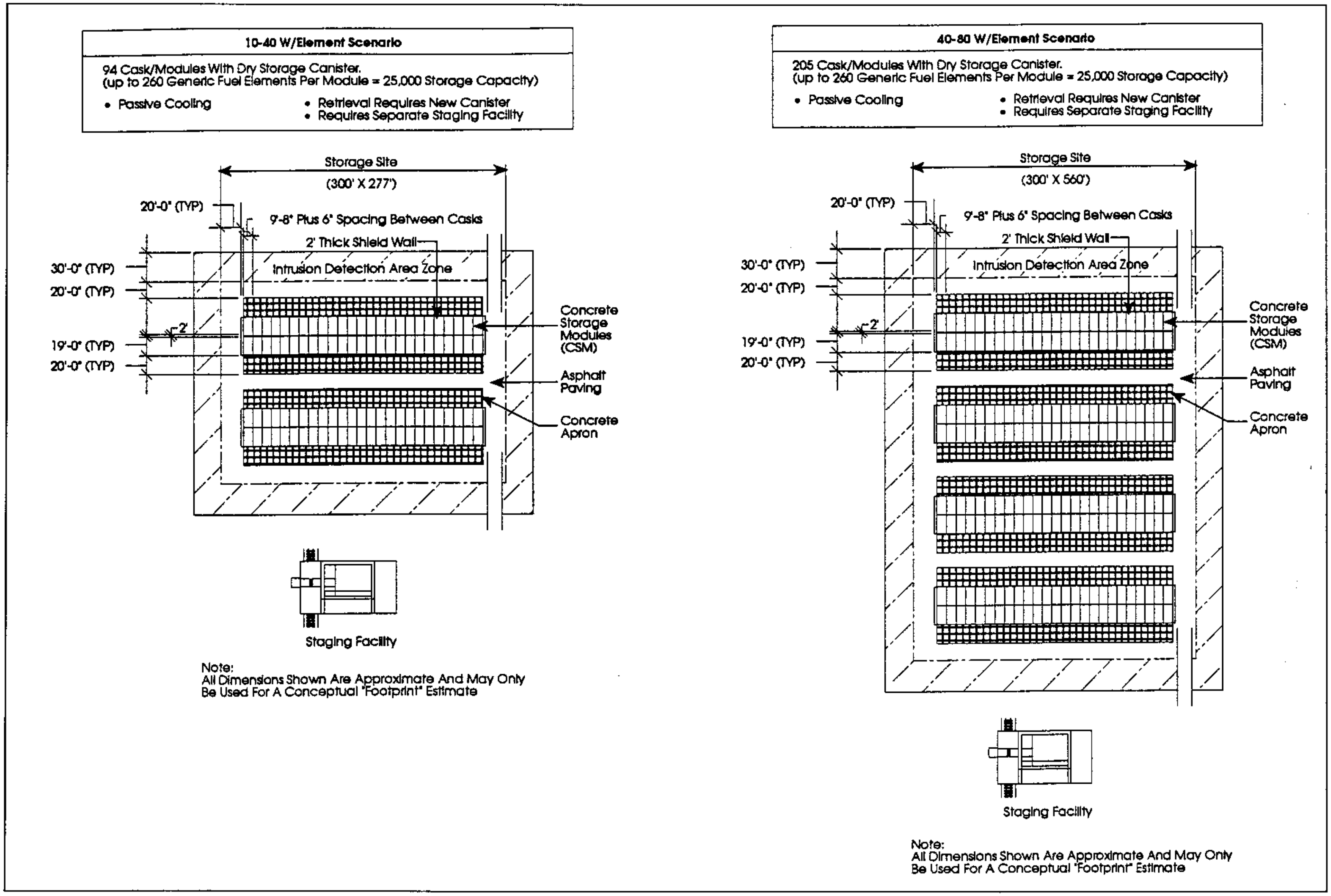


Figure F-35 Schematic of Expanded Dry Cask Storage Facility (10 to 40 Watts Scenario and 40 to 80 Watts Scenario)

Table F-18 Summary of Dry Cask Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel[a]

Construction Phase:	
Disturbed Land Area	3 ha (7.7 acres)
Facility: Size (Area) Concrete Steel	 2,200 m^2 (24,000 ft^2) 17,500 m^3 (22,900 yd^3) 4,500 metric tons (5,000 tons)
Soil Moved	11,000 m^3 (14,400 yd^3)
Equipment Fuel	810,000 l (214,000 gal)
Construction Debris/Waste	1,800 m^3 (2,400 yd^3)
Work Force	50/yr for staging facility, 50 per 25 cask array, 1 array/yr
Duration (Years)	4 years for construction, 1.5 years for design
Capital Cost	$366 million[b]
Operation Phase:	
Electricity	800 - 1,000 MW-hr/yr (staging facility)
Water	2.1 million l/yr (550,000 gal/yr) during receipt, 0.9 million l/yr (238,000 gal/yr) thereafter
Wastestreams	
Solid Low Level Waste	16 m^3/year (565 ft^3/year) during receipt, 1 m^3/yr (35 ft^3/yr) thereafter
Waste Water	1.58 million l/yr (412,000 gal/year) during receipt, 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30 during receipt, 8 thereafter
Annual Operating Cost	$17.3 million during handling, $0.3 million during storage[b]

[a] *Staging facility parameters based upon the Regionalized, Small Wet Pool (Dahlke et al., 1994)*

[b] *Cost estimates are in $1993 (EG&G, 1993)*

to mitigate the environmental effects of any radionuclide releases. This facility is normally located immediately adjacent to, or within, the staging facility.

Dry cask storage is unique among the three storage technologies because of its ability to be operationally integrated with existing facilities, which allows for faster implementation as compared to the other two storage technologies. Several DOE sites have facilities with spent nuclear fuel handling capabilities similar to the requirements of the staging facility. Potential examples include the RBOF at the Savannah River Site and the ICPP-666 storage pool area. For dry cask storage, the spent nuclear fuel would be shipped to the existing facility and unloaded from the transportation cask. The spent nuclear fuel would be inspected, canned if identified as a degraded element, and placed inside the storage canister. Spent nuclear fuel with heat loads exceeding 40 Watts per element would be stored in the existing facility to allow cooldown prior to cask storage. After filling, the canister would be sealed and placed inside the storage cask. The only new construction required would be the concrete storage pad (for vertical casks) or the concrete storage modules (for horizontal casks). For the foreign research reactor spent nuclear fuel receipt rate of approximately 2,000 elements per year considered in the analyses in this EIS, approximately 8 storage casks would be needed annually.

The cost to construct a dry cask storage facility with a staging area sufficient to unload, characterize, can, temporarily store in a small pool, and transfer the spent nuclear fuel to the cask storage area is estimated to be $366 million. The annual operating cost for this facility is estimated to be $17.3 million during the period of handling and transfers of the spent nuclear fuel and $0.3 million during the period of storage.

The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993) with the addition of the cost of a small wet storage facility reported by Dahlke et al. (Dahlke et al., 1994).

F.3.2 Wet Storage Facility

Three generic wet storage facility options have been proposed for foreign research reactor spent nuclear fuel. They are denoted Centralized-Underwater Fuel Storage Facility, Regionalized Large-Underwater Fuel Storage Facility, and Regionalized Small-Underwater Fuel Storage Facility (Dahlke et al., 1994). The difference between these 3 options is that Centralized-Underwater Fuel Storage Facility is sized to store 100 percent of the foreign research reactor spent nuclear fuel under consideration in this EIS (Figure F-36), Regionalized Large-Underwater Fuel Storage Facility is designed for the storage of 75 percent of the foreign research reactor spent nuclear fuel, and Regionalized Small-Underwater Fuel Storage Facility will accommodate 25 percent of the foreign research reactor spent nuclear fuel. These three options were selected to encompass any conceivable decision regarding centralization or regionalization (by geography or fuel type) for the foreign research reactor spent nuclear fuel storage sites. The design features of all three wet storage facility options are identical with the exception that building and pool sizes and, in the case of the Regionalized Small-Underwater Fuel Storage Facility, the number of storage pools and receiving bays is smaller for the Regionalized Large-Underwater Fuel Storage Facility and Regionalized Small-Underwater Fuel Storage Facility. Table F-19 presents the difference in design between these three facilities. Because the design and environmental impacts of the larger Centralized-Underwater Fuel Storage Facility would bound the two smaller facility designs, the balance of the presentation in this section addresses the specific design of the Centralized-Underwater Fuel Storage Facility for storage of 100 percent of the foreign research reactor spent nuclear fuel.

The proposed new wet storage facilities consist of a fuel storage area and support areas (Dahlke et al., 1994). The Fuel Storage Area provides for the receipt of cask transportation vehicles, cask unloading and decontamination, fuel handling, transfer, and storage. Support areas provide for the equipment necessary to maintain and operate the storage area (e.g., heating, ventilation, air conditioning, water treatment, and waste management). The wet storage facility would be constructed as a structure that meets all current nuclear regulations for withstanding natural events such as seismic, tornado, and flood, as well as aircraft impact loads. All systems supporting the operation of the fuel storage facility would also meet these safety requirements. The facility is equipped with a 118-metric ton (130-ton) overhead cask handling crane, and a 9-metric ton (10 ton) fuel handling crane. Each cask transportation vehicle would enter the facility through one of two bays, where it would be monitored and washed from transportation dust. When the external surfaces are cleaned, the cask would be placed into a decontamination room where the cask would be prepared as needed to facilitate underwater unloading. The cask would then be placed in an unloading pool. Transportation casks would be monitored and, if clean of radioactive contamination, placed in an unloading pool. The cask receiving area can accept two simultaneous shipments on 3 m (10 ft) by 24.4 m (80 ft) trucks or railcars and casks weighing up to 114 metric tons (126 tons), each with a total individual cask and transport vehicle weight of 177 metric tons (195 tons).

There are two stainless steel-lined unloading pools, one measuring 6.4 m (21 ft) long, 5.8 m (19 ft) wide by 13.4 m (44 ft) deep, and the other measuring 6.1 m (20 ft) long, 6.1 m (20 ft) wide, and 11 m (36 ft) deep. There are two decontamination hot cells. Each unloading pool has a cask washdown system. Prior to being placed in one of the two storage pools, each fuel element would be checked to ensure that it is properly configured for direct transfer to the fuel storage pool buckets. If not, it would be transferred to the fuel cutting/canning pool, which is 10.4 m (34 ft) long, 5.8 m (19 ft) wide, and 9.4 m (31 ft) deep. Here it would be prepared for transfer to the storage pool buckets.

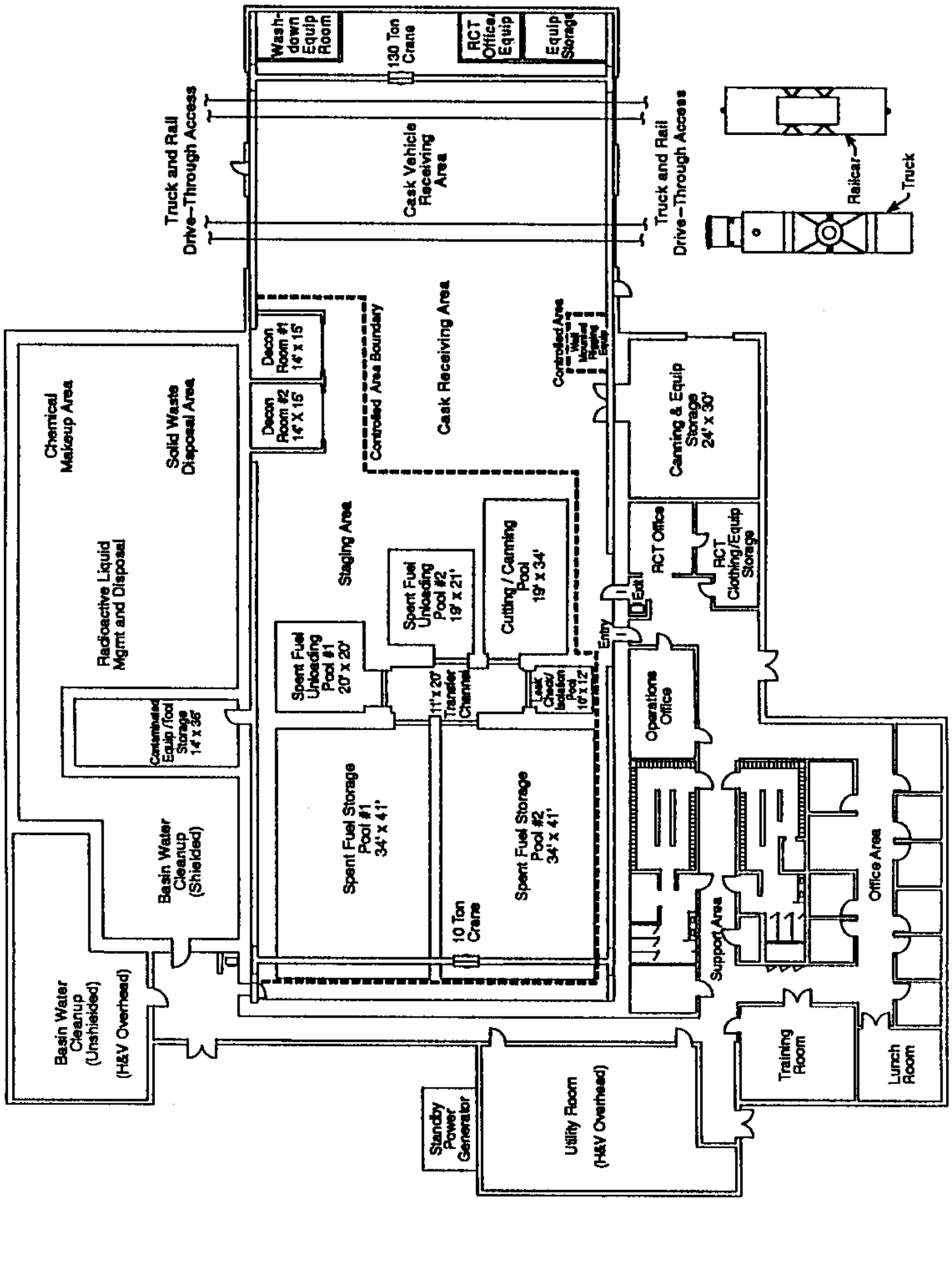


Figure F-36 Generic Wet Storage Facility for All of the Foreign Research Reactor Spent Nuclear Fuel

Table F-19 Design Difference Between 100 Percent, 75 Percent, and 25 Percent Generic Wet Storage Facilities

Design Parameter	Wet Storage Capacity (Amount of Foreign Research Reactor Spent Nuclear Fuel)		
	100%	75%	25%
Number of Storage Pools	2	2	1
Storage Pool Length and Width, m (ft)	16.5 x 10.4 (54 x 34)	12.5 x 10.4 (41 x 34)	10.4 x 8.2 (34 x 27)
Transfer Channel Length and Width, m (ft)	6.1 x 3.4 (20 x 11)	6.1 x 3.4 (20 x 11)	6.1 x 3 (20 x 10)
Fuel Unloading Pool Length and Width, m (ft)	6.4 x 5.8 (21 x 19)	6.4 x 5.8 (21 x 19)	6.1 x 6.1 (20 x 20)
Number of Receiving Bays	2	2	1

If cask measurements indicate that fuel is degraded, the fuel would be transferred to the isolation pool which is 3.7 m (12 ft) long, 3 m (10 ft) wide, and 9.4 m (31 ft) deep. This pool is equipped so that wet sipping, dry sipping, or vacuum sipping of the suspect fuel element could be performed. Sipping is a method of measuring radioisotope leakage from spent nuclear fuel. An identified degraded fuel element would then be transferred to the cutting/canning pool, where it would be canned before transfer to the storage pool. If it is not found to be degraded, it would be transferred directly to the storage pool.

All six pools in this facility (two unloading; two storage, cutting/canning; and two leak check/isolation) are hydraulically connected by a stainless steel-lined transfer channel/pool which is 6.1 m (20 ft) long, 3.3 m (11 ft) wide, and 9.4 m (31 ft) deep. Gates between this transfer channel and each pool allow for hydraulic watertight isolation of the other pools to control contamination and allow for individual pool water pump-out. All pools and channels are constructed of concrete with stainless steel floors and liners. Pool water leak detection and collection systems are provided in accordance with NRC Regulatory Guide 1.13 (NRC, 1975) and American National Standards Institute Standard N305-1975 (ANSI, 1975b).

Each of the two stainless steel-lined, interconnected storage pools is 16.5 m (54 ft) long, 10.4 m (34 ft) wide, and 9.4 m (31 ft) deep. Each contains stainless steel storage racks which hold 1,000 fuel storage holes, with a 0.2 m (8 in) spacing maintained between adjacent storage holes (Figure F-37). Each storage hole can hold three stacked stainless steel fuel storage buckets (Figure F-38), which can each contain up to four fuel elements. A loading fixture is used during spent nuclear fuel emplacement. Thus, each pool has the capacity for 12,000 fuel elements (or 24,000 for both pools). The 0.2-m (8-in) space provides neutron isolation between adjacent storage holes and, therefore, ensures criticality safety. Each rack is 2 m (6.7 ft) square and 3.2 m (10.4 ft) high, and consists of a 5 x 5 array of 25 fuel positions. A hinged lid is above each of these fuel positions. Each pool can hold 40 of these racks. Fuel elements are stored in the racks so that at least 25.4 cm (10 in) of rack protrudes above the top of the fuel.

The heating, ventilation, and air conditioning system for the wet storage facility would include a room for air supply equipment and a room for air exhaust equipment with separate filtering and monitoring rooms for the different areas within the facility. There are two heating, ventilation and air conditioning equipment rooms. Although a total of six different rooms are used for heating, ventilation, and air conditioning, all exhaust air is directed through pre-filters, High Efficiency Particulate Air filters, radiation monitors, filter fire protection components, and heat recovery coils before it exhausts to the atmosphere.

The wet storage facility's water treatment system consists of redundant pumps, piping, filters, deionizers, and ultraviolet microorganism control systems. A heat removal system is sized to maintain the bulk water temperature at or below 43°C (109°F). The system's filters and deionizers include anion and cation exchangers that maintain water chemistry and remove radionuclides from the pool water.

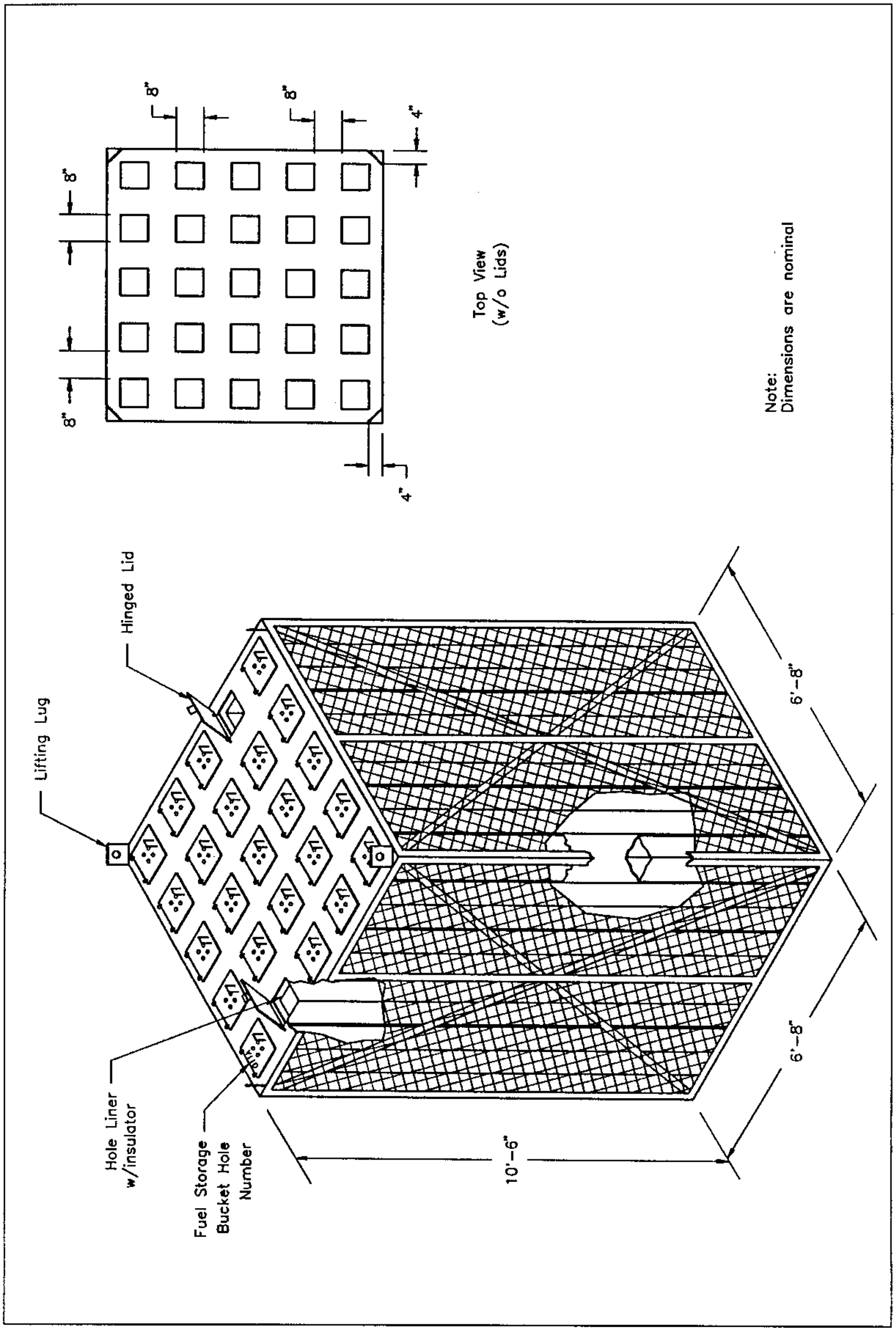


Figure F-37 Storage Racks

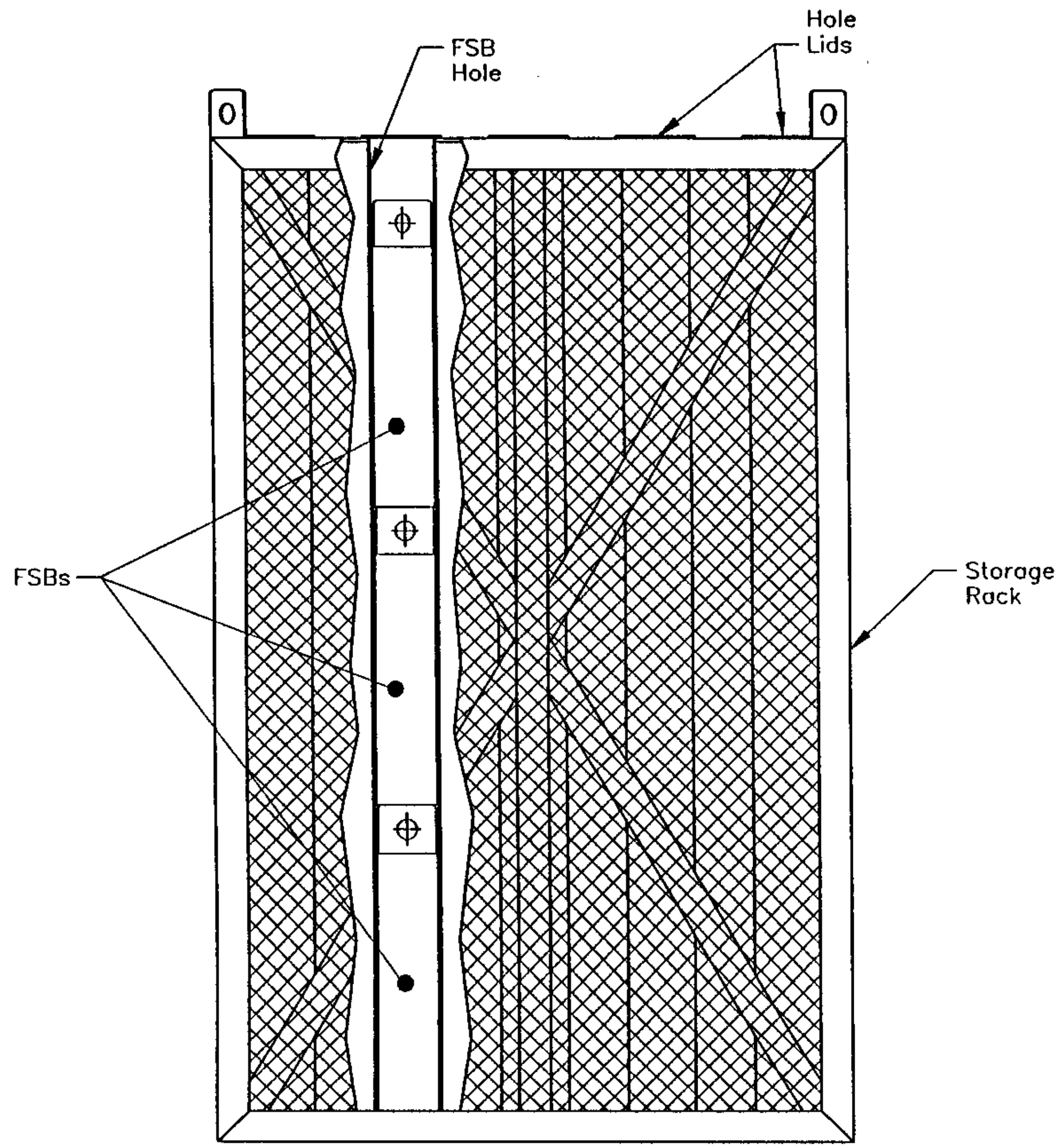


Figure F-38 Stacked Fuel Storage Buckets in Storage Rack

The staff required to operate the wet storage facility is estimated to be a maximum of 30 when 24-hour-a-day fuel loading is being performed, with only occasional maintenance visits by administrative personnel for operation. Potential radiological consequences are extrapolated from other operating wet storage facilities and are discussed in Section F.4.

No high-level radioactive waste is expected to be generated by the wet storage facility. Low-level solid radioactive waste generated over the 40-year life of the facility is expected to be about 488 m^3 (17,200 ft^3). Nonradioactive solid waste generated over the facility's life is expected to be about 300 m^3 (10,594 ft^3). No nonradioactive air emissions are expected to be generated by this facility. Table F-20 summarizes the parameters for the facility.

The cost to construct a wet storage facility with a staging area sufficient to unload, characterize, can, and transfer the spent nuclear fuel to the storage area is estimated to be $449 million. This cost may include some duplicate facilities and equipment present in both the staging facility and the rest of the wet storage facility which were costed separately. The annual operating cost for this facility is estimated to be $23.3 million during the period of handling the spent nuclear fuel and $3.5 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993).

F.3.3 Site-Specific Facilities Proposed for Foreign Research Reactor Spent Nuclear Fuel Storage and Management

F.3.3.1 Savannah River Site

RBOF

The Savannah River Site has proposed the use of its RBOF, which is also designated as Building 244-H (DuPont, 1983a and 1983b; Shedrow, 1994a and 1994b; WSRC, 1994a; Claxton et al., 1993; DOE 1993c). The RBOF is a 30-year-old steel and concrete block structure that contains several water pools that have been used for the storage of spent nuclear fuel including foreign research reactor spent nuclear fuel since approximately 1964.

The RBOF facility is located in H-Area on 0.8 ha (2 acres) of land about 397 m (1,300 ft) west of the 221-H Canyon building. A railroad track terminates within the facility, and a roadway surrounds it for access by trucks. The RBOF Building (244-H) is about 42 m (139 ft) wide and 45 m (148 ft) long and contains water-filled basins. The basin area extends below grade to a maximum depth of 13.7 m (45 ft), the roof over the 91 metric ton (100 ton) crane bay is about 13.7 m (45 ft) above grade, and most of the remainder of the roof is at an elevation of 4.6 m (15 ft).

The building consists of seven main sections separated by partition and shielding walls. A ventilation system is provided to exhaust any airborne particulate contamination through filters. The basins, cubicles, and shielding walls are made of reinforced concrete. Most of the above-grade structure consists of standard structural steel shapes with an exterior wall of Transite™ (registered trademark of Johns-Manville Co.). The walls are insulated with Fiberglas™ (registered trademark of Owens-Corning Corp.).

The basin, or working area, of the building has an inner wall of Transite™ to prevent water damage to the insulation from condensation. The disassembly and inspection basins are separated by an inner concrete block wall, and the repackaging basins are enclosed by concrete block walls.

Table F-20 Summary of Wet Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel

Construction Phase:	
Disturbed Land Area	2.8 ha (7 acres)
Facility: Size (Area) Concrete Steel	 3,800 m^2 (41,000 ft^2) 12,400 m^3 (16,260 yd^3) 3,100 metric tons (3,443 tons)
Soil Moved	18,000 m^3 (24,000 yd^3)
Equipment Fuel	600,000 l (159,000 gal)
Construction Debris/Waste	2,600 m^3 (10,300 yd^3)
Work Force	157/yr average, 184 peak
Duration (Years)	4 years for construction, 1.5 years for design
Capital Cost	$449 million[a,b]
Operation Phase:	
Electricity	1,000 - 1,500 MW-hr/yr
Water	2.7 million l/yr (720,000 gal/yr) during receipt, 1.5 million l/yr (409,000 gal/yr) thereafter
Wastestreams	
High Level Waste	none
Transuranic Waste (TRU)	none
Solid Low Level Waste	16 m^3/yr (580 ft^3/yr)
Waste Water	1.59 million l/yr (420,000 gal/yr) during receipt, 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30
Annual Operating Cost	$23.3 million during handling $3.5 million during storage[a]

Source: (Dahlke et al., 1994)

[a] *Cost estimates are in $1993 (EG&G, 1993)*

[b] *The cost may include duplicate equipment costed in both the staging facility and the wet storage facility*

The two storage pools are between 6.7 and 8.8 m (22 and 29 ft) deep, and approximately 70 percent of their capacity is filled with a variety of fuel types, including aluminum-clad fuel with a ^{235}U enrichment up to 93.91 percent. Subcriticality is maintained by appropriate fuel spacing [center-to-center fuel spacing in these racks currently varies between 23 and 65 cm (9 and 25.5 in) depending on the specific rack] in the storage racks, since no neutron absorption material is used in the pool water. Rack height is 3.4 m (11.17 ft), but some fuel protrudes above the top of the racks. Some of this fuel has been stored at the RBOF for as long as 15 years without any significant degradation. These aluminum storage racks have been present in the pools for 30 years without degradation. Figure F-39 shows the floor plan, and Figure F-40 displays an elevation view.

The RBOF includes specific design, operating, and maintenance procedures for the receipt of a wide variety of fuel types and casks, including damaged fuel elements. The RBOF has the facilities and experience in all aspects of spent nuclear fuel receipt including cask wash, fuel unloading, fuel transfer, fuel storage, fuel inspection, fuel disassembly, and fuel repackaging.

The RBOF pools have a stainless steel bottom and epoxy-coated walls. Pool walls are made of reinforced concrete that varies in thickness from 0.9 m (3 ft) at the top of the pool to 2 m (6.5 ft) at the lower elevations, and the pool floor is a stainless steel liner over a 91-cm (3-ft) thick reinforced concrete slab. Most of the pools have a 6.4-mm (0.25-in) thick stainless steel liner on the floor. The disassembly, inspection, and repackaging basins also have a 3.2-mm (0.125-in) thick stainless steel liner on the walls. Pools or basins are connected by transfer canals with underwater doors.

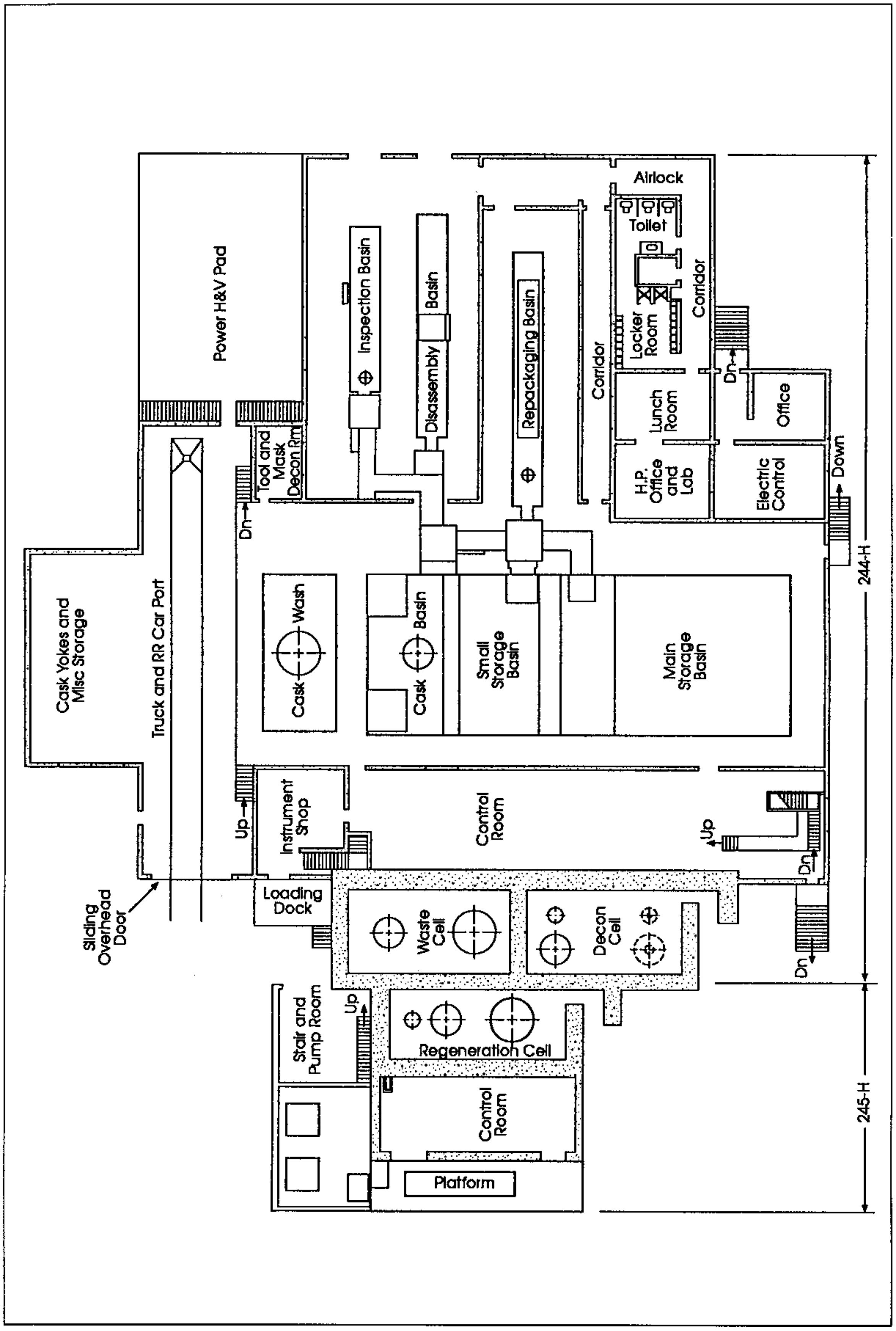


Figure F-39 Plan View of the RBOF

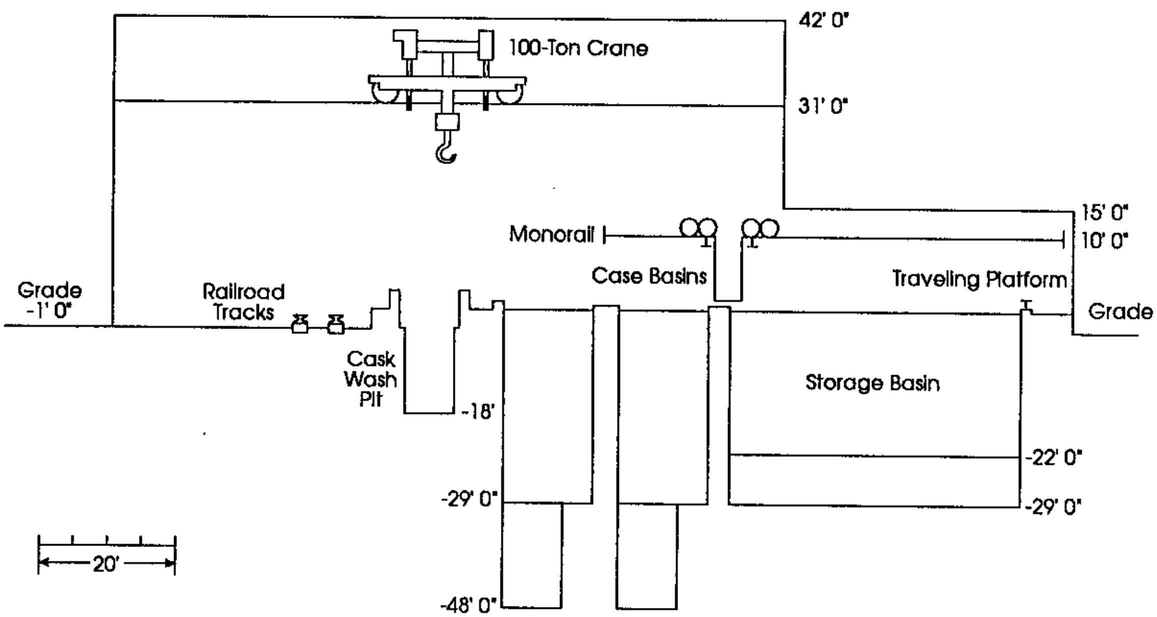


Figure F-40 Elevation Schematic of the RBOF (Facing East)

Water from any basin can be pumped through a filter-deionizer and then returned to the basin as purified water with a conductivity in the range of 0.5 to 1.5 μmhos/cm. In addition, the activity level of the water, which is typically in the range of 0.5 to 1.5 x 10^{-4} μCi/ml, is reduced to less than 0.05 x 10^{-4} μCi/ml by this process. The normal inventory of activity in the approximately 1,700,000 l (450,000 gal) of total basin water is thus 0.1 to 0.3 Ci. For typical flow rates of 454 l/min (120 gal/min), the deionizer processes approximately 1.1 x 10^{8} l (3 x 10^{7} gal) during 6 months of service and may contain about 20 Ci of radioactivity (mostly as cesium) when regeneration is required. The deionizer is typically regenerated every 4 to 5 months. The "Porostone" (aluminum oxide) filter that precedes the deionizer is normally backflushed and recoated with filter-aid whenever a significant pressure drop occurs, which, in practice, is about three to four times per month. When the filter tubes become plugged, they are chemically treated with oxalic acid and sodium hydroxide to open the pores of the filter. This occurs once or twice per year. The purification system maintains excellent chemistry, with mercury and copper kept below two parts per billion, and iron and aluminum maintained below 2 parts per million (ppm). Chloride is maintained below 10 parts per billion.

In the event of an interruption of normal power to the RBOF, critical equipment essential for maintaining personnel safety and containing radioactivity are automatically supplied with emergency power from a 12.5-kilovolt amps, 10-kilowatts, 460-volts gasoline-driven generator.

The building ventilation system serves to minimize airborne radioactivity both inside and outside the facility because of the "once-through" airflow system and the use of High Efficiency Particulate Air filters on the building exhaust. Because of the once-through airflow, activity levels do not tend to increase with time within the building, and the High Efficiency Particulate Air filters serve to effectively remove particulate radioactivity that would otherwise be released to the atmosphere. In addition, the facility is maintained at less than atmospheric pressure so that all building air will be filtered before release. The basin areas are also kept at a lower pressure than the rest of the building, and a separate ventilation system supplies air to the control room, offices, and change room. All air, after filtration, is discharged through a 1.5-m (5-ft) diameter by 16.2-m (53-ft) high stack. The exhaust system for the process vessels in the Waste Cell and the Decontamination Cell is similar to the building system, but is separate and employs acid resistant components. This exhaust is discharged through a 25-cm (10-in) diameter by 16.2-m (53-ft) high pipe.

The 91-metric ton (100-ton) capacity bridge crane travels on a 27.4-m (90-ft) long runway located 9.4-m (31-ft) above grade which permits access to the carport, the cask wash pit, and the cask basin. It is used to handle transportation casks, cask lids, cask basin shims, and a semi-remote impact wrench.

The twin hook crane consists of two 45-metric ton (50-ton) capacity hoist trolleys, which can be arranged for independent travel, or which can be electrically locked to provide for operation as a single unit. Load clearance above the 1.1-m (3-ft 6-in) high cask basin railing is 8.1-m (26-ft 6-in). The bridge crane is pendant-operated from a walkway on the west side of the basins.

A 2.7-metric ton (3-ton) hoist, suspended from a monorail on the south girder of the bridge, is used in the handling of a semi-remotely operated impact wrench. The other 2.7-metric ton (3-ton) hoist is used primarily for the handling of yokes and other ancillary equipment in the yoke storage area adjoining the carport.

Brakes on the cranes and hoist are applied automatically in the event of a power outage.

Two small bridge cranes, one motorized and one manually operated, are employed over the repackaging basin. Both have a load capacity of 2.7 metric tons (3 tons).

The RBOF includes a High Efficiency Particulate Air heating, ventilation, and air conditioning system and maintains subatmospheric pressure within the building to minimize environmental releases of radionuclides. Automatic atmospheric isolation is actuated by activity level monitors inside the RBOF. The RBOF is also equipped with groundwater monitoring for detecting leakage from the pool confinement boundary.

Analysis of the RBOF was performed and included an evaluation of the reliability of process equipment and controls, administrative controls, and engineered safety features. The evaluation identified potential scenarios and radiological consequences. Risks were calculated in terms of 50-year population dose commitment per year (person-rem per year) to the onsite staff and to an individual at the plant boundary. Risk is defined as the product of the expected frequency of a release and the consequences of the release. Consequences are expressed in terms of dose commitment to onsite and offsite populations surrounding the release point.

An evaluation of the RBOF as a potential storage site for foreign research reactor spent nuclear fuel indicates a number of problem areas. The current cask handling capacity of the RBOF is approximately one cask per week. This capacity is based upon facility operations at two shifts per day, 5 days per week. The cask handling capacity could be increased, perhaps to as much as 84 casks per year, if facility operations were expanded to around-the-clock (3 shifts per day), 7 days per week. However, considering that shipments out of the RBOF also require cask handling, the net receipt capacity of the RBOF is practically limited to four casks per month. This capacity would not be sufficient for the potential foreign research reactor spent nuclear fuel cask receipt rate of ~60 casks per year. If the RBOF were used for the receipt and loading of dry storage canisters, its receipt rate could be reduced by half. Only ~1,000 fuel storage spaces are available at the RBOF. Consolidation of the spent nuclear fuel might open an additional 1,425 spaces, but this is much less than that required for the number of foreign research reactor spent nuclear fuel elements under consideration in this EIS. The Savannah River Site has proposed movement of other spent nuclear fuel to the reactor storage basins, and use of dry storage for foreign research reactor spent nuclear fuel.

The DOE Spent Fuel Working Group Report has identified a number of vulnerabilities at the RBOF, including insufficient training, inadequate tornado missile protection, no seismic qualification, lack of water leak detection system, and no up-to-date and approved Safety Analysis Report (DOE, 1993b). It should be noted that a system description and a Safety Analysis Report for the RBOF do exist and were published in 1983. Current recommendations are to address and correct these problems by FY 1996 (DOE, 1993b; Taylor et al., 1994). The 30-year age of these pools may also require analyses to determine the remaining safe lifetime without significant replacement or design modifications.

Reactor Disassembly Basins

Savannah River Site has also proposed the use of one or more of its reactor disassembly basins for Phase 1 storage of foreign research reactor spent nuclear fuel (Shedrow, 1994a and 1994b; Taylor et al., 1994). All of these basins were constructed in the early 1950's and became operational in the mid-1950's. The disassembly basins are similar to each other and are briefly described in the sections that follow, using the L-Reactor disassembly basin as an example.

The L-Reactor performed the basic function of irradiating elements in a heavy water moderated and cooled reactor for the purpose of supplying special nuclear materials for national defense, medical, and research applications. The Savannah River Site production reactors are not currently operating. The disassembly area of the Savannah River Site Production Reactors was designed to serve as a processing area for reactor target and fuel assemblies. This processing included removal of decay heat, disassembly of components, short term storage of fissile product material, and cask loading operations. Total residence time from reactor discharge to shipment to the separation areas was typically 12 to 18 months.

The disassembly basin is arranged into three major sections: the machine basin, the vertical tube storage basin, and the transfer area (Figure F-41). The machine basin and vertical tube storage basins are divided into the following interconnected basins:

Vertical Tube Storage Basin	*Machine Basin*
Deposit and Exit Canal*	Machine Area
Vertical Tube Storage	Horizontal tube storage Target bucket storage Dry cave

* *The Deposit and Exit canal is a water-filled canal that connects the disassembly area and the area that houses the reactor tank top or process room. This canal also acts as a water seal to allow access by an underwater conveyer, but no airflow.*

The disassembly basin contains 12,776,000 l (3,375,000 gal) of water. Vertical tube storage holds 3,730,000 l (985,000 gal), while the machine basin holds 9,050,000 l (2,390,000 gal) (transfer area included). The depth of the disassembly basin ranges from 5.2 to 9.2 m (17 to 30 ft), but most of the basin is around 5.2 m (17 ft) deep. The approximate overall dimensions are 47 m (154 ft) wide by 66 m (216 ft) long. There is a small 0.9 m (3 ft) diameter circular section of the machine basin that is 15.6 m (50 ft) deep.

Figure F-42 shows a basic block diagram of the disassembly process. The reactor assemblies were discharged from the reactor tank to the Deposit and Exit canal, placed in the Deposit and Exit conveyer,

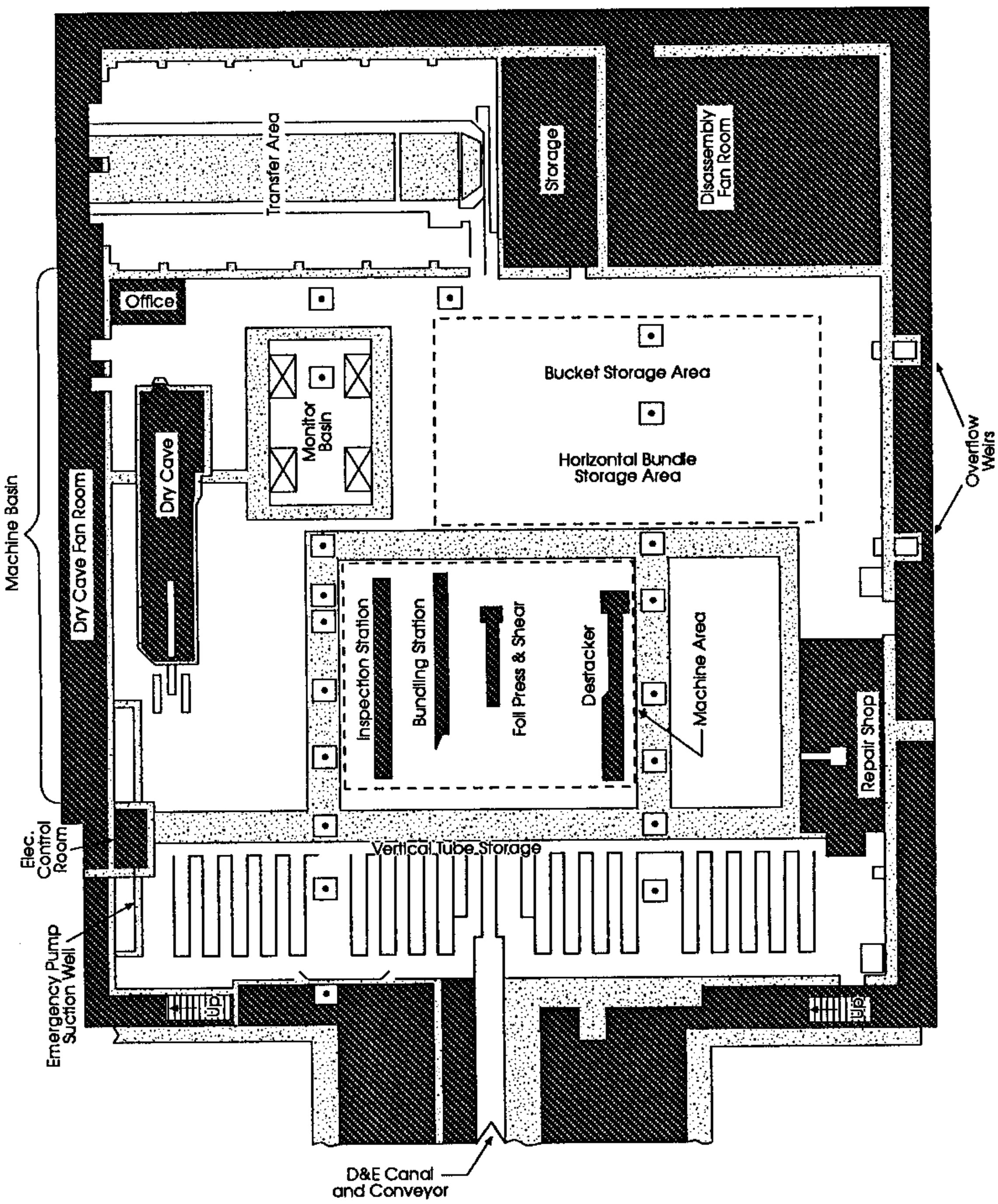


Figure F-41 Typical Reactor Disassembly Basin Area Layout

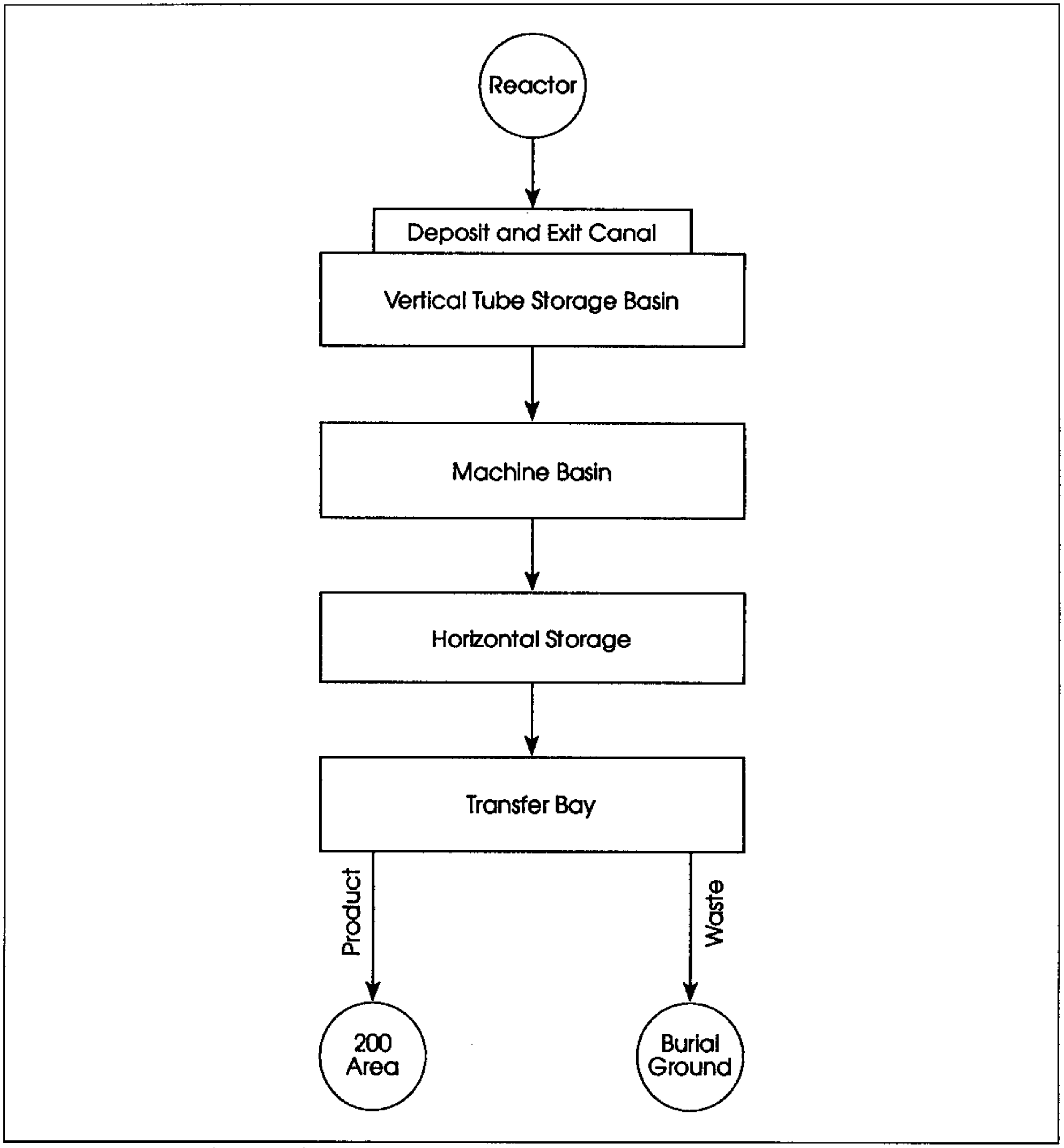


Figure F-42 Reactor Disassembly Basin Process Block Diagram

and transferred to the disassembly area side of the Deposit and Exit canal. The assemblies were transferred to hangers suspended from overhead monorails and were initially stored in the vertical tube storage area for 3 to 8 months. Fuel and target assemblies were moved to the machine basin area where they were disassembled. Target material was placed in stainless steel buckets and then stored in the bucket storage area. Fuel was bundled in aluminum bundles and stored in horizontal storage racks. The components were then allowed to cool for up to another 8 months. The fuel and target material was expected to be in the basin no longer than 18 months. The components, once sufficiently cooled, were moved to the transfer area to be shipped by cask to the processing facility.

Use of a disassembly basin for foreign research reactor spent nuclear fuel would require continuous demineralizer treatment for water quality, and new storage racks. Existing heat exchanger systems can remove upwards of 6,800 kilowatts (24 x 10^6 BTU/hr), which should far exceed the 240-1,000 kilowatts heat generation rate of foreign research reactor spent nuclear fuel (i.e., 10 to 40 Watts per element). These changes would allow each basin to accommodate approximately 20,000 elements.

The transfer area provides an area for shipping or receiving material and equipment. This area consists of two water-filled basins which are designated the scrap pit and the transfer pit. Irradiated material ready for shipping is transferred from horizontal storage to the transfer bay. Transportation casks are moved to and from the transfer pit and irradiated material placed in the cask using hoists mounted on the monorail system. Transportation casks can be transported to and from the reactor areas by tractor trailer or railroad. Trailers or railcars are positioned inside the transfer pit and casks are lifted and transported into/out of the basin using an 85/30 ton overhead crane.

Over the course of the site's history, at least 10 different casks have been used for various applications, many of which are still available for use pending proper inspection and maintenance. Two types are now used for most, if not all, disassembly work. EP-85 is a 63.5-metric ton (70-ton) fuel and target transport cask and EP-383 is a 13.6 metric ton (15-ton) cask used to move scrap to the burial ground.

The transfer area cranes would have to be modified to accommodate the different casks used for offsite shipments. These changes would allow a disassembly basin to receive up to seven casks per month in addition to the projected Savannah River Site shipping requirements.

A monorail system is mounted to the ceiling throughout the disassembly area. This system is used to transport and store all types of spent nuclear fuel and reactor components in the disassembly basin area. Most of the disassembly monorail system is designed for a working load of 907 kg (2,000 lbs) per foot of rail, which would be adequate for moving foreign research reactor spent nuclear fuel.

H-Canyon

The H-Area facilities occupy approximately 160 ha (395 acres). The H-Area Canyon processed irradiated fuel elements via modifications of the plutonium-uranium extraction process (Figure F-43) and is oriented toward HEU recovery. Primary operations also include the dissolution of fuel tubes, chemical and physical separations, and purification of materials. DOE stores the high-level waste from the operations in large tanks (nominally, 3.8 million l or 1 million gal each) for future stabilization and disposal via the Defense Waste Processing Facility.

The facility arrangement of the H-Canyon is identical to the F-Canyon. The main facility in the H-Area is the 221 H-Canyon, where most of the separations of irradiated materials were accomplished. The H-Canyon is a Class 1 reinforced concrete structure with exterior dimensions of 254.4 m (835 ft) by 37.2 m (122 ft) by 20.1 m (66 ft) high. The facility houses two parallel process canyons, each 9.1 m (30 ft) wide. The two process canyons handle high activity and lower activity materials, and are designated as the hot and warm canyons, respectively. Each of these process canyons is divided into 14 process cells, 13.1 m (43 ft) long by 4.6 m (15 ft) wide by 13.7 m (45 ft) high. Each process canyon is serviced by an overhead crane that operates the entire length of the canyon. The cranes perform remote operations for the processes such as equipment replacement, piping and electrical changes, leak repair, and inspections. Recently, new cranes were installed in both canyons. These cranes are remotely operated and include a Closed Circuit Television system for better monitoring capabilities. The warm process canyon crane has a lifting capacity of 27.2 metric tons (30 tons) and the hot process canyon's crane capacity is 45.4 metric

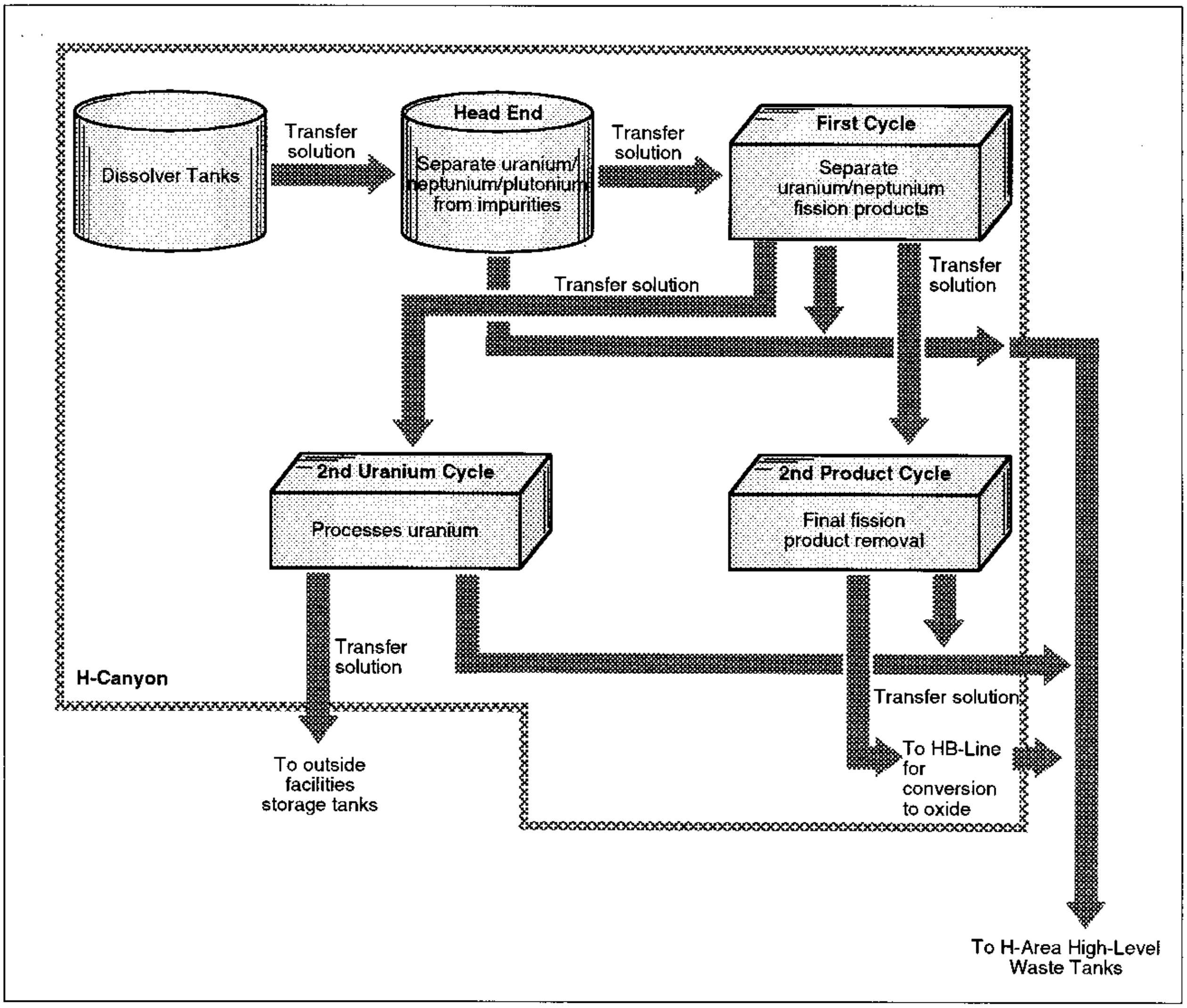


Figure F-43 H-Canyon Modified Plutonium-Uranium Extraction Process Flow

tons (50 tons). A separate maintenance area for each canyon provides for crane maintenance and repair. The original canyon cranes are still in place and can be utilized as backups to the new ones.

Spent nuclear fuel is delivered by a water-filled railway mounted cask into a shielded tunnel. The hot canyon overhead crane unloads spent nuclear fuel either into a storage basin in Section 3 or into the dissolver. The H-Canyon basin was never designed for storage of spent nuclear fuel, but only as a staging area prior to reprocessing. As such, it does not have the capability to establish, circulate, or maintain water chemistry. Basin water is sampled and analyzed every 3 months and domestic water is added manually every 3 months by opening an appropriate valve to the basin to make up for evaporation.

The H-Canyon storage basin is constructed of concrete and is 5.4 m (17.75 ft) long, 2.38 m (7.83 ft) wide, and 7.6 m (25 ft) deep. The floor is covered by a stainless steel liner that also extends 4.3 m (14 ft) up each wall. A stainless steel storage rack sits in the bottom of the basin and provides adequate spent nuclear fuel separation. The basin is filled with water to a level of 3.8 m (12.5 ft). The fuel currently being stored in the basin contains approximately 40 kg (88 lbs) of ^{235}U. The H-Canyon is currently in a status ready for restarting. An EIS was prepared to cover the canyon's restart for processing of the liquids currently stored in tanks within the facility. Future missions for the facility are still being analyzed. For foreign research reactor spent nuclear fuel, H-Canyon could be used for chemical separation and blending down of HEU to LEU material.

F.3.3.2 Idaho National Engineering Laboratory

On October 17, 1995, litigation with the State of Idaho was settled by stipulation of the parties and entry of a consent order. This settlement would provide for the transportation of up to 61 shipments of foreign research reactor spent nuclear fuel to the Idaho National Engineering Laboratory prior to the year 2000, if DOE and the Department of State choose to adopt a policy of accepting such spent nuclear fuel. After the year 2000, additional shipments of such spent nuclear fuel could be made to the Idaho National Engineering Laboratory under the stipulated settlement and consent order. However, the following discussion is to provide a full understanding of the existing capabilities at the Idaho National Engineering Laboratory in light of the fact that this site has been considered as a reasonable alternative to manage the foreign research reactor spent nuclear fuel as in the preparation of the Draft EIS.

Irradiated Fuel Storage Facility

The ICPP-603 includes an underwater fuel storage basin area and the IFSF, which is a remotely operated, dry-vault facility specifically constructed for the storage of graphite fuel from the Fort St. Vrain and Peach Bottom reactors. It was built in 1974 as an addition to the underwater Fuel Storage Facility and contains 636 storage positions. This facility can handle casks weighing up to 55 metric tons (60 tons). Spent nuclear fuel currently stored here is from two commercial high-temperature, gas-cooled reactors (Fort St. Vrain and Peach Bottom), some for the ROVER Nuclear Rocket Program, and some Tory 2C and BER II TRIGA fuel. The IFSF is a good candidate for spent nuclear fuel requiring frequent monitoring because of the ease of visual fuel inspections.

Since the facility can accommodate fuels up to 3.0 m (130 in) in length, all types of foreign research reactor spent nuclear fuel under consideration in this EIS could be handled. Transfer cart modifications would be needed for the proposed foreign research reactor spent nuclear fuel transportation casks, since the existing transfer cart only has the capability to handle the Rover cask, the Fort St. Vrain cask, and the Peach Bottom cask. New fuel handling tools, such as a new can grapple, would be needed. New cell preparations and work stations would also be needed. Sipping, unloading, canning, sealing, and leak checking equipment would need to be added to the cell. A 14 metric ton (15 ton) crane is present in the vault room for fuel handling. Visual inspection and gamma spectroscopy could readily be performed in the existing vault room.

The vault room is 7 m by 7.1 m by 6.6 m (22 ft 10 in by 23 ft 3 in by 21 ft 6 in) high, and the storage room is 437 m^2 (4,700 ft^2). Loaded fuel cans can be transferred from the vault room to the storage area by a shuttle bin. The vault room is being reanalyzed structurally to validate its capability to meet the current seismic requirements of 10 CFR 72. Recent reliable data regarding the effectiveness of the filtering and ventilation systems must be obtained in order to assess the amount of radionuclides that may be vented into the outside air. The cost to add the required capabilities to the IFSF for storage of foreign research reactor spent nuclear fuel is approximately $5 million.

It should be mentioned that, although this facility was originally constructed to accommodate the Fort St. Vrain High Temperature Gas-Cooled Reactor graphite fuel, it will not be used for this purpose because of the October 16, 1995 Settlement Agreement with the State of Idaho that declared that the Fort St. Vrain fuel will not be brought to the Idaho National Engineering Laboratory for interim storage. Public Service of Colorado has obtained a 10 CFR 72 license to store all of the fuel in the Foster Wheeler modular dry vault facility built adjacent to the reactor site. That modular dry vault is currently completely loaded with Fort St. Vrain fuel, and the reactor is being decommissioned. Availability of the IFSF for foreign research reactor spent nuclear fuel will be dependent on decisions to consolidate spent nuclear fuel from other Idaho National Engineering Laboratory facilities at this facility.

Approximately 300 positions are available in the IFSF dry storage facility for foreign research reactor spent nuclear fuel. Preparations to receive the foreign research reactor spent nuclear fuel could be completed as soon as calendar year 1997. However, many activities are already scheduled for this facility. A new canning station for spent nuclear fuel from the ICPP-603 basins is being constructed in the handling cave, and the canning will then be accomplished. Other spent nuclear fuel management activities being considered in this facility include ROVER reactor fuel shipments, Experimental Breeder Reactor II and FERMI movements fuel from ICPP-666. Naval fuel inspection sample receipts from the Expended Core Facility are scheduled. The Peach Bottom fuel in the ICPP-749 facility and Rover fuel ash transfers from a shutdown fuel processing facility are also being considered for repackaging in the IFSF handling cell, and the Idaho National Engineering Laboratory spent fuel consolidation activities are scheduled to begin within 2 years. Detailed facility usage schedules have been drafted to demonstrate how the foreign research reactor spent nuclear fuels could be accommodated in this workload.

With this schedule as background, foreign research reactor spent nuclear fuel preparations could take place during early 1996. Fuel could be received beginning in 1997 (61 shipments through FY2000) and up to 101 shipments in the years beyond this timeframe.

The 300 positions could store up to approximately 9,000 foreign research reactor spent nuclear fuel elements. Approximately 60 dry storage positions in the ICPP-749 drywells could also be utilized to store foreign research reactor spent nuclear fuel. Some refurbishment would have to take place to receive spent nuclear fuel. This could be completed in 1997, and the facility would be ready to receive fuel at the beginning of 1998. With 60 fuel elements per position, up to 3,600 fuel elements could be stored in the ICPP-749. Any fuel that is stored in the ICPP-749 would have to go through the ICPP-603-IFSF to be placed in sealed containers and transferred to an interfacility transfer cask.

Idaho Chemical Processing Plant-666 Fuel Storage Area

This facility (Figure F-44) is the modern Idaho National Engineering Laboratory underwater storage facility. Receipt and storage of foreign research reactor spent nuclear fuel has been accomplished in the past as one of its many missions. It has the capability of receiving and unloading spent nuclear fuel casks at a rate of approximately five per week. Storage capability for up to 8,400 foreign research reactor spent nuclear fuel elements can be provided for an approximate 10-year period by using the increased capacity fuel storage racks that will be installed in Pool 1 via a reracking project planned following the Programmatic SNF&INEL Final EIS, and by installing additional fuel storage racks in the cutting pool. The increased capacity being provided in Pool 1 will be required for Naval spent nuclear fuel receipts beginning in about FY 2005. The racks being removed from Pool 1 as part of the reracking project could be placed in the cutting pool for the foreign research reactor spent nuclear fuel.

The capability of the ICPP-666 facility to receive foreign research reactor spent nuclear fuel in the near term is limited, however, due to the number of activities scheduled through 1998. These activities include reracking in Pools 1, 6, and 5. Fuel receipts from the Navy and the Advanced Test Reactor, and fuel transfers from CPP-603 will continue. The CPP-603 fuel transfers are required to meet a court order requirement. These activities utilize nearly all resources, such as cranes, manpower, and health physics personnel in the near term of 1995 through 1997. A limited number of shipments (one per month) are possible for the 1996 and 1997 schedules. In 1998, the schedule relaxes enough that up to 30 shipments

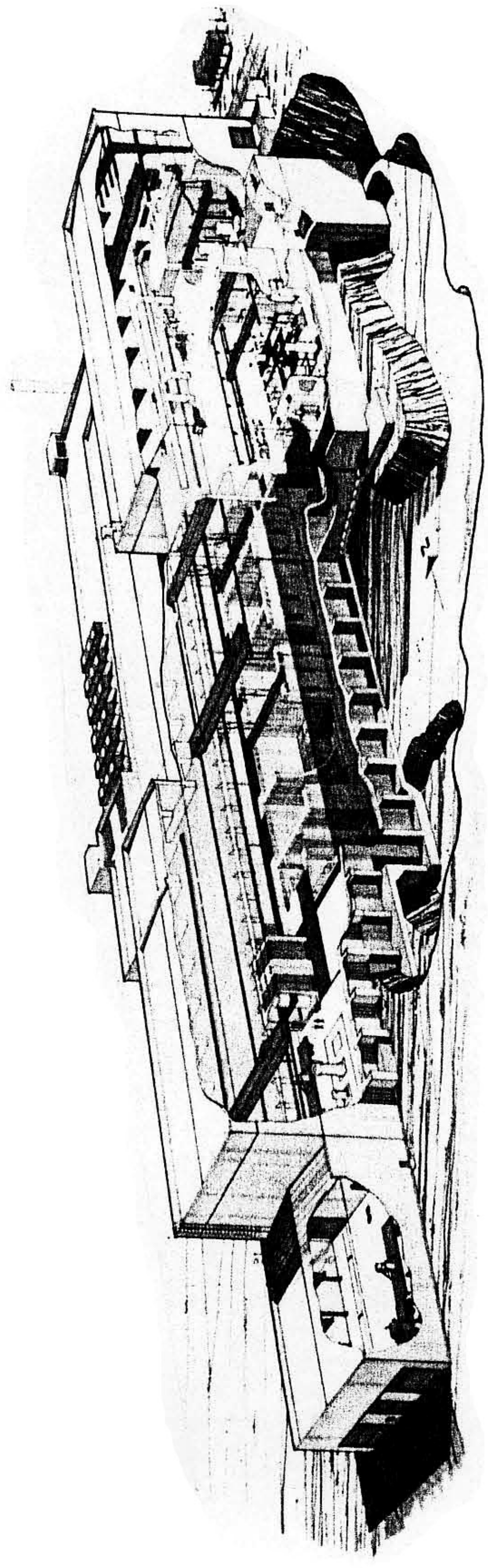


Figure F-44 Pictorial of the Idaho Chemical Processing Plant-666 could be received because most of the ICPP-603 fuel transfers to Fluorinel Dissolution and Fuel Storage (FAST) will have been completed. By the end of 1999, a total of 3,600 fuel elements could be received under this scenario. This schedule is predicated on the assumption that resolution of Idaho National Engineering Laboratory facility vulnerabilities and Naval and Advanced Test Reactor fuel receipts have a higher priority than foreign fuel receipts.

Idaho Chemical Processing Plant-666 Fluorinel Dissolution Process

The conversion of the ICPP-666 Flourinel Dissolution Process cell for canning of spent nuclear fuel without removal of any existing equipment or decontamination of the cell is proposed as a low-cost option to prepare foreign research reactor spent nuclear fuel for dry storage. Only minor modifications would be made to the cell, so as to preserve the current dissolution capability for possible future use. This option would utilize the Flourinel Dissolution Process cell, which currently has remote fuel handling, sampling, and waste load-out capabilities, as well as a connection to the ICPP-666 Fuel Storage Area, for fuel inspection, stabilization, and packaging for interim dry storage.

A potential disadvantage is noted. The equipment inside the Flourinel Dissolution Process cell is contaminated, and radiation fields are too high for manned entry. Retaining the dissolution cell equipment will make it impossible to adequately clean the cell to allow personnel entry. For this reason, equipment requiring installation within the Flourinel Dissolution Process cell must be assembled outside the cell and installed remotely with the in-cell crane and master-slave manipulators. Preventative and corrective maintenance of the equipment inside the cell would be done remotely. The modular design of the components would facilitate removal and replacement.

No general Flourinel Dissolution Process utility upgrades, such as electrical power or ventilation, would be required. Piping services could be added to support the vacuum drying and inert gas backfilling functions envisioned to meet dry storage requirements. All of the Reduced Enrichment for Research and Test Reactors (RERTR) program spent nuclear fuel could be accommodated by the existing transfer tunnel and transfer cart. Modifications of the existing fuel shear tools, or new ones, could be acquired to shear larger objects such as cans and lids.

Fuel Processing Restoration

Another structure that may represent an option for fuel storage is the Fuel Processing Restoration building (Figure F-45) that was constructed to house the Fuel Processing Restoration process. It is approximately 56.4 m (185 ft) long and contains shielded, below-grade process cells. These cells vary in dimension, with the nine main process cells measuring 5 to 6 m (16.5 to 20 ft) wide, 10.4 m (34 ft) long, and 12.2 m (40 ft) deep. Fuel racks could be designed to accommodate cans of the type proposed for dry storage in arrays that could contain as many as 17 cans along the 10.4 m (34 ft) axis by 8 cans along the 4.9 m (16 ft) axis, and be 3 cans deep. Airflow through each of these cells could be controlled by dampers in the cell ductwork. Construction of this facility was interrupted prior to completion, and it currently does not include cell ventilation, fire safety equipment, instrumentation, or lighting. These additions would cost approximately \$15 million. Several other processes are being considered for use of this facility, and there is no assurance that it would be used for fuel storage. When completed, the building will be seismically qualified, and could physically accommodate approximately 540 canisters per main process cell. Special remote handling tools and techniques would be developed to allow the fuel cans to be inserted into the storage cell. The total estimate to make all necessary conversions is \$65 million.

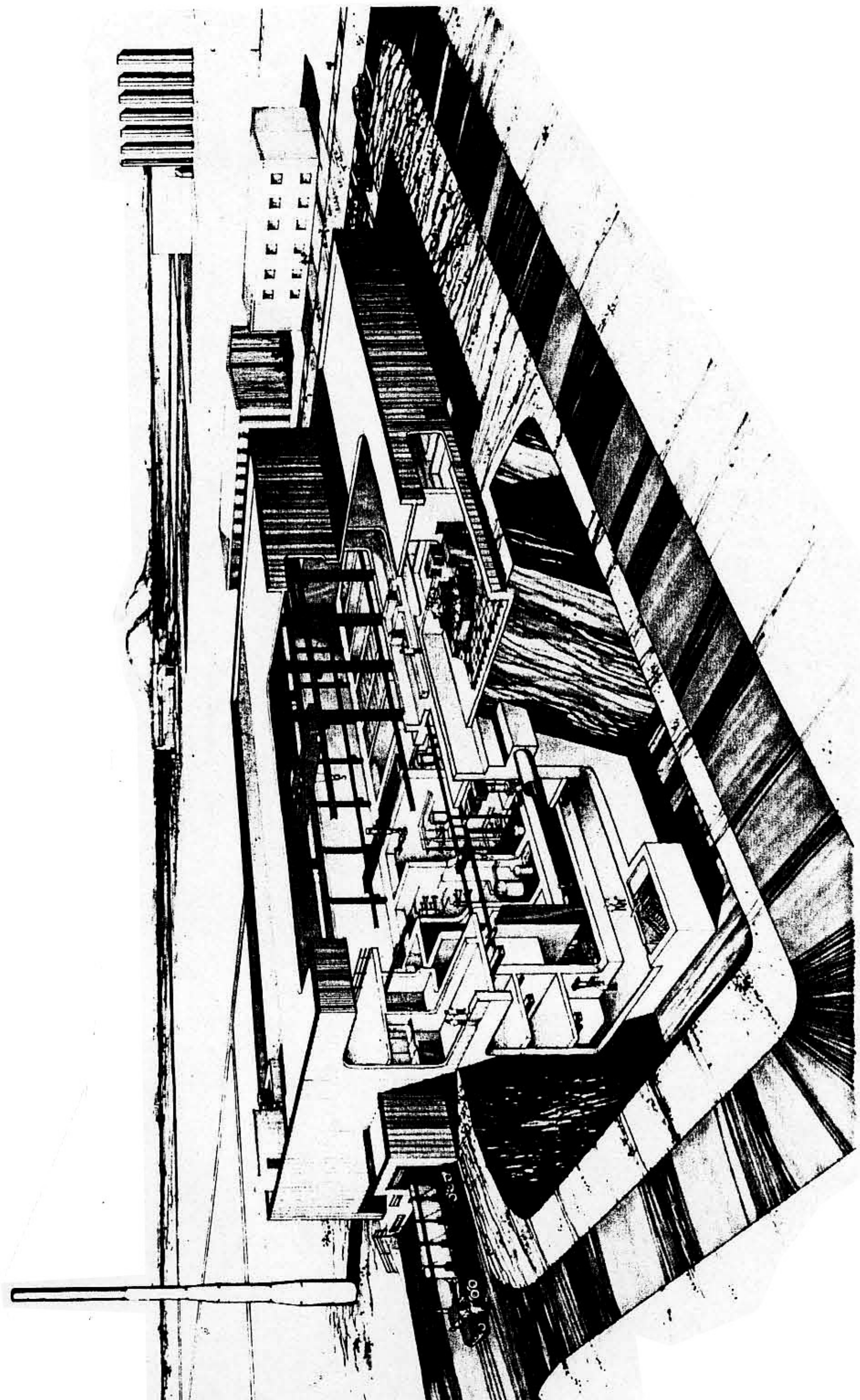


Figure F-45 Fuel Processing Restoration Facility (Unfinished)

Hot Fuel Examination Facility

The Hot Fuel Examination Facility is a facility used for examining and storing irradiated fuels from the EBR-II breeder reactor at Idaho National Engineering Laboratory. Figure F-46 presents the layout of the main cell of the Hot Fuel Examination Facility. Although it was not designed for storage, the Hot Fuel Examination Facility could be used to receive, inspect, examine, and transfer foreign research reactor spent nuclear fuel to dry cask storage if it is fitted with an appropriate spent nuclear fuel examination station. The cost of these modifications and the purchase and installation of the dry casks and their equipment would be the principal costs involved.

Test Area North-607 Pool, Hot Cell, and Cask Storage Pad

The utilization of the Test Area North-607 facilities is a potential option for receipt and storage of the foreign research reactor spent nuclear fuel. This could be accomplished without significant modification to the hot cell. The hot cell has significant lag capacity for interim storage of foreign research reactor spent nuclear fuel and has most of the equipment necessary for placement of the fuel into dry interim storage casks. There is adequate space for installation of the characterization and conditioning equipment needed for dry storage. There are significant vulnerabilities associated with the underwater storage pool which would need to be corrected if underwater interim storage were desired. The cask storage pad could be easily expanded to accept additional dry storage casks. At the current time, the entire Test Area North area is being planned for shutdown in approximately ten years due to reduced mission needs. If Test Area North had adequate new missions and it was determined to be economical, the Test Area North hot cell and cask storage area would have significant capacity for receipt and temporary storage of foreign research reactor spent nuclear fuel.

F.3.3.3 Hanford Site

In addition to the generic dry and wet storage facilities, two existing facilities at Hanford Site have been identified as potential candidates for the storage of foreign research reactor spent nuclear fuel: the FMEF and the Washington Nuclear Plant-4 Spray Pond. They will be discussed in greater detail in the following sections.

F.3.3.3.1 Fuel Maintenance and Examination Facility (FMEF)

The FMEF, built during the late 1970s and early 1980s (but never completed), consists of a 82.3 m (270 ft) long, 53.3 m (175 ft) wide, 29.9 m (98 ft) high Process Building with attached mechanical equipment and entry wings. The FMEF was intended to receive and extensively examine irradiated breeder reactor test fuels. The seismically-qualified FMEF Process Building, which extends 10.7 m (35 ft) below the surface, consists of 6 operating floors or levels and encloses a total of 17,466 m^2 (188,000 ft^2) of operations space. The FMEF has a 68 metric ton (75 ton) overhead crane, 18 metric ton (20 ton) hoist, and 9 metric ton (10 ton) hoist. Three areas within the FMEF, the Shipping and Receiving area, Decon Cell, and the Entry Tunnel would be used for the storage of foreign research reactor spent nuclear fuel. Figure F-47 presents a ground floor plan, and Figure F-48 shows vertical storage of spent nuclear fuel canisters.

The Shipping and Receiving Area, also known as Room 300, is the access area to the FMEF for truck or rail shipments of spent nuclear fuel. This area provides a 24.4 m (80 ft) long working area, washdown and decontamination of casks and shipping vehicles, and crane interface access to other areas within the FMEF. The 68 metric ton (75 ton) crane and hoists can transport casks to the Entry Tunnel from this area. The principal modification needed for this area would be the construction of a 0.8 km (0.5 mi) rail extension to the existing Hanford Site rail system.

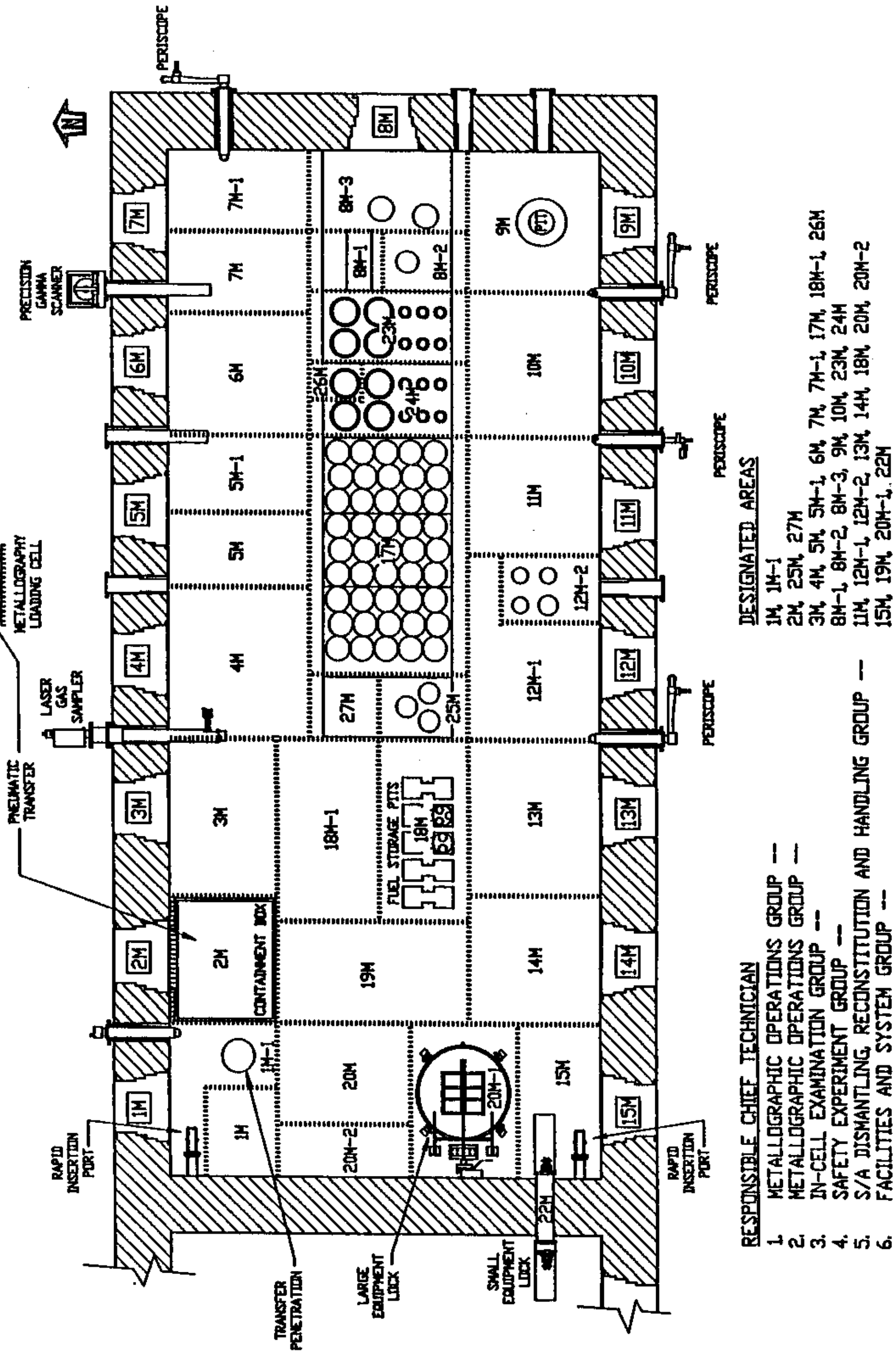


Figure F-46 Hot Fuel Examination Facility Main Cell Layout

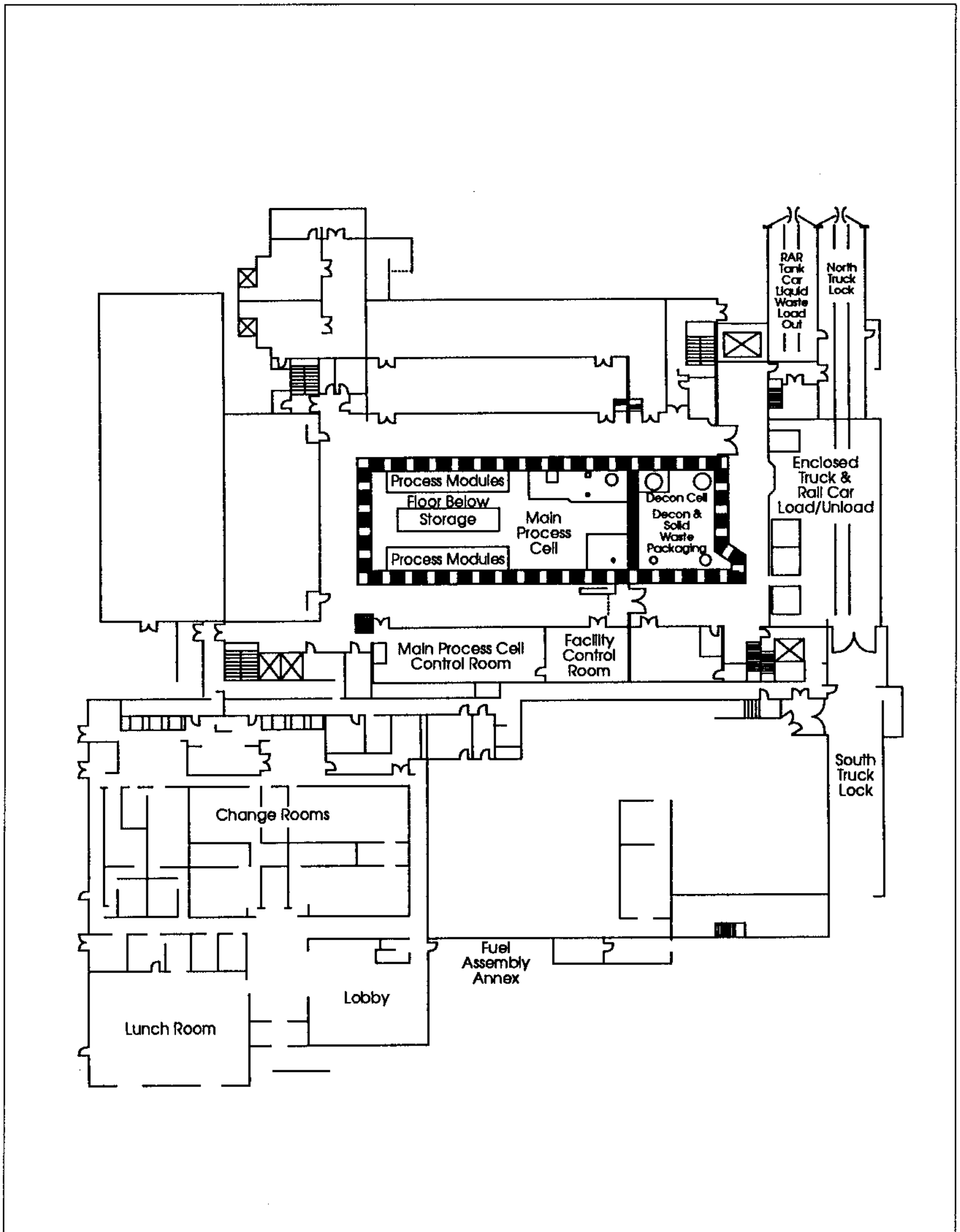


Figure F-47 Fuel Maintenance and Examination Facility 0'-0" Level Floor Plan

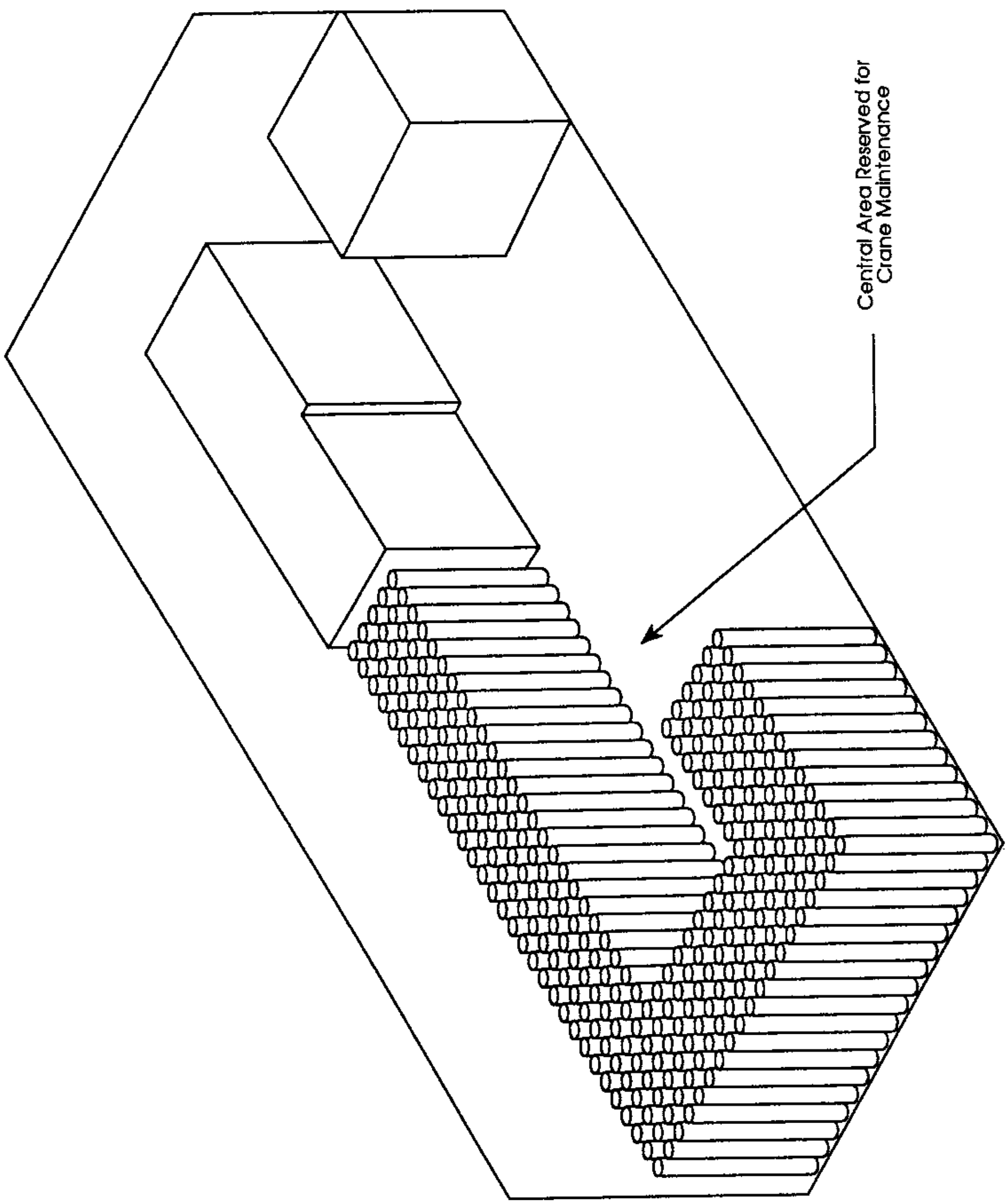


Figure F-48 Potential Use of the Fuel Maintenance and Examination Facility for Fuel Storage Partial Plan View

The Entry Tunnel is a 9.9 m (32.5 ft) high tunnel below the Shipping and Receiving area floor designed to transfer transportation casks to the Decon and Main Process Cell areas of the FMEF. It includes a 68 metric ton (75 ton) overhead bridge crane. To transport heavier multi-purpose casks when the foreign research reactor spent nuclear fuel would be shipped out of the FMEF in the future, the tunnel would be extended and modified to accommodate an above-grade 114 metric ton (125 ton) crane. The entry tunnel would be extended to connect to the new adjacent storage facility.

The Decon Cell is a room 12.2 m (40 ft) long, 9.1 m (30 ft) wide, and 11.6 m (38 ft) high, with thick concrete shielding. Nine work stations with remote manipulators and viewing windows are part of the design of this cell, although it should be noted that the viewing windows and equipment have not been installed. Access is available through two 2.1 m (84 in) diameter hatches, a 0.8 m (30 in) diameter port, and a 0.3 m (12 in) diameter opening. The Decon Cell includes material handling capability by cranes, manipulators, and hoists ranging from 1.4 to 6.8 metric tons (1.5 tons to 7.5 tons). The Decon Cell would be used to inspect foreign research reactor spent nuclear fuel that has been unloaded from transportation casks and to subsequently load this spent nuclear fuel into storage baskets. Each basket holds three foreign research reactor spent nuclear fuel elements. After basket loading, four baskets would be stacked into a 4.6 m (15 ft) high stainless steel canister. The canisters would be moved to the adjacent storage facility using a Transfer Tunnel, which is equipped with a cart.

The Main Process Cell represents a potential storage location for foreign research reactor spent nuclear fuel at the FMEF (Figure F-47). This room is 30.5 m (100 ft) long, 12.2 m (40 ft) wide, and 11.6 m (38 ft) high with concrete walls either 1.2 or 1.5 m (4 or 5 ft) thick, depending on the concrete's density. The Main Process Cell design includes two 4.5 metric ton (5 ton) bridge cranes and two 1.4 metric ton (1.5 ton) electro-mechanical manipulators.

A zoned heating, ventilation, and air conditioning system with negative differential pressure, redundant cooling systems, and staged multiple High Efficiency Particulate Air filters provides decay heat removal and protection from environmental releases of radioisotopes. This system provides for flow, by negative air pressure differential, from the least contaminated zones to the most contaminated zones, thereby maintaining individual zone relative contamination potential. Supply air is drawn from tornado-hardened and seismically qualified intake shafts and dampers. All heating, ventilation, and air conditioning equipment required to supply high contamination zones is designed as Seismic Category 1. After multiple High Efficiency Particulate Air filtration and monitoring for radioactivity, heating, ventilation, and air conditioning exhaust air is released from a seismically qualified reinforced concrete 35.7 m (117 ft) tall stack.

The FMEF is provided normal power by two separate 115 kV electric power supply lines from the Bonneville Power Administration. Emergency power is provided by two 100 percent redundant 900-kilowatt gas turbines which, along with their seismically qualified support and fuel oil systems, are capable of 24 hours of continuous operation. An Uninterruptible Power Supply, consisting of two 150-kVA lead calcium batteries, can provide full load for 30 minutes. Emergency generators require 2 minutes to start up and produce rated power.

A number of modifications would be required for the FMEF to be used as a storage facility for foreign research reactor spent nuclear fuel. They can be categorized as: addition of a 114 metric ton (125 ton) crane, railroad tunnel extension, and storage rack canisters. Even with these modifications, the FMEF does not have sufficient space to store 23,000 foreign research reactor spent nuclear fuel elements, but it could be used as an unloading and support facility for an adjacent dry vault storage facility. Costs for the necessary modifications to the FMEF have been estimated to be approximately $32 million. The adjacent dry storage facility is estimated to cost an additional $100 million. It should be noted that the FMEF is being considered for other spent nuclear fuel storage which could eliminate it for use with foreign research reactor spent nuclear fuel.

F.3.3.3.2 Washington Nuclear Plant-4 Spray Pond Wet Storage

The Washington Nuclear Plant-4 Spray Pond is a nuclear safety-related structure that was originally designed for decay heat removal following a Loss of Coolant Accident at the Washington Nuclear Plant-4 commercial nuclear power plant. The Washington Nuclear Plant-4 was canceled, but the spray pond structure is essentially complete. This pond is 91.4 m (300 ft) long, 76.2 m (250 ft) wide, and 8.2 m (27 ft) deep, and was designed and built to 10 CFR 50 Appendix B quality assurance standards as a seismic and safety class structure. It should be noted that the size of this spray pond is much greater than that needed to store the foreign research reactor spent nuclear fuel. Figure F-49 presents a schematic of the Washington Nuclear Plant-4 Spray Ponds and the necessary modifications required to store foreign research reactor spent nuclear fuel.

In order for this pond to be used for wet storage of foreign research reactor spent nuclear fuel, several modifications would have to be made to duplicate the features of the Generic Pool Facility. These modifications include: an enclosure with a qualified building superstructure, inclusion of a shipping-receiving-handling facility in this structure; a deeper loading pool; a 114 metric ton (125 ton) loading crane; a 274 m (900 ft) long railroad line extension; installation of heating, ventilation, and air conditioning, and pond water cooling and water chemistry/cleaning systems; canister support racks into the pond; installation of a stainless steel liner in the pond; installation of partition walls to isolate the fuel storage and handling areas from the unused portion of the spray pond; and a leak detection system. The supporting pumphouse is a 12 m (40 ft) by 30 m (100 ft) concrete building designed to the same standards as the spray pond and would be used to house some of the fuel handling and storage support equipment. Total cost for these modifications was estimated to be approximately $113 million (Bergsman et al., 1994).

F.3.3.4 Oak Ridge Reservation

No existing facilities at the Oak Ridge Reservation are being used for the storage of foreign research reactor spent nuclear fuel. However, either the generic wet (pool) or dry storage facilities, as described in Sections F.3.1 and F.3.2, could be constructed and operated at Oak Ridge Reservation.

F.3.3.5 Nevada Test Site

No existing facilities at the Nevada Test Site are being used for storage of foreign research reactor spent nuclear fuel, although the Area 25 facilities (E-MAD/Reactor Maintenance and Disassembly) have been used in the past and might be suitable in 1 to 3 years. These Area-25 facilities appear capable of accommodating the required cask receipt rate and dry storing all of the foreign research reactor spent nuclear fuel under consideration in this EIS. However, in addition to Area 25, either the generic wet (pool) or dry storage facilities, as described in Sections F.3.1 and F.3.2, could be constructed and operated at Nevada Test Site in Area 5.

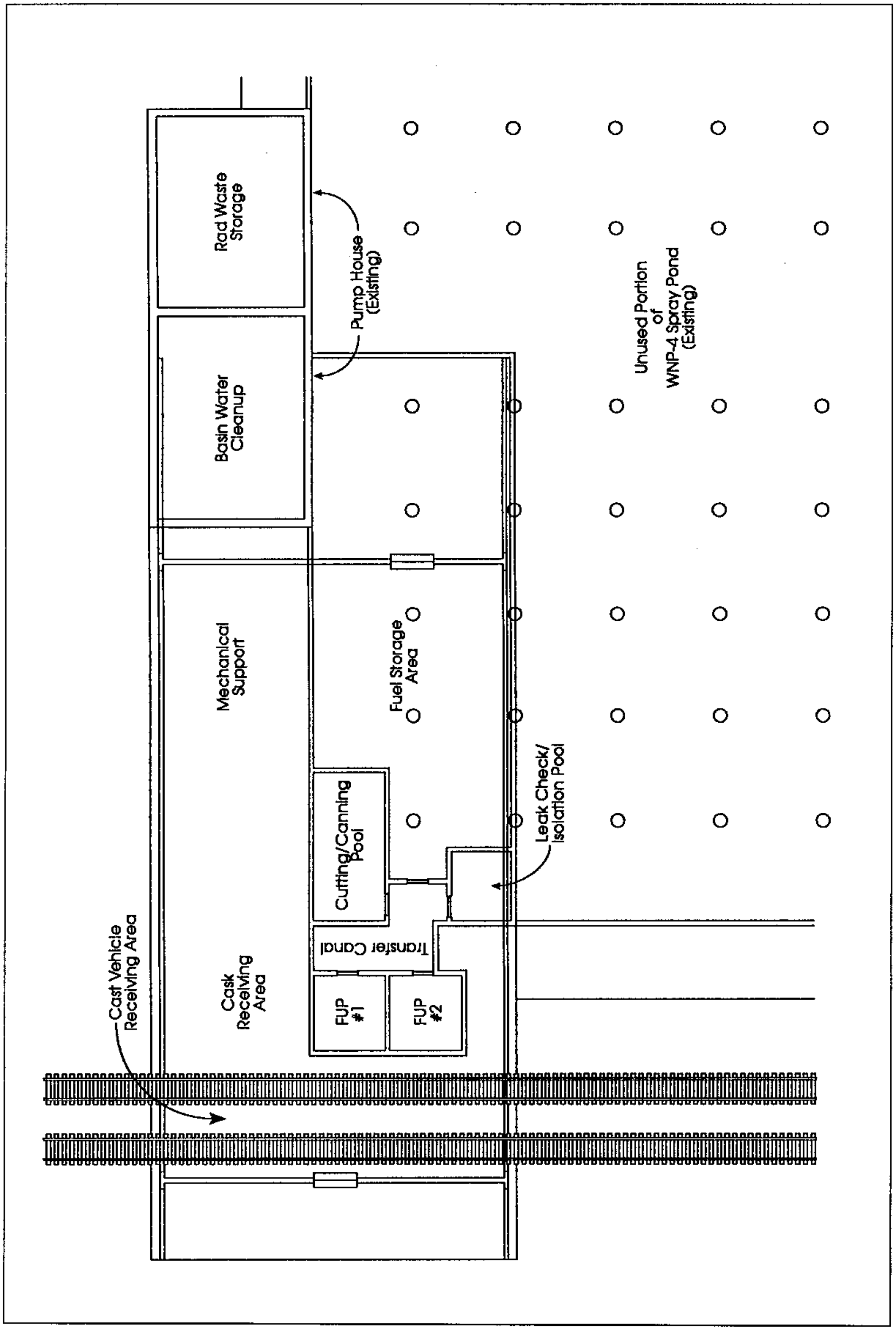


Figure F-49 Washington Nuclear Plant-4 Spray Pond (with Modifications) Schematic

F.4 Environmental Impacts at Foreign Research Reactor Spent Nuclear Fuel Management Sites

This section analyzes the environmental impacts associated with the storage of foreign research reactor spent nuclear fuel at the five potential management sites considered in the Programmatic SNF&INEL Final EIS (DOE, 1995g), namely: the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site.

The Record of Decision for the Programmatic SNF&INEL Final EIS was issued on May 30, 1995. In accordance with this Record of Decision, all of the aluminum-based foreign research reactor spent nuclear fuel accepted by DOE would be managed at the Savannah River Site in South Carolina, and any other foreign research reactor spent nuclear fuel to be accepted by DOE would be managed at the Idaho National Engineering Laboratory. Nevertheless, all five of the spent nuclear fuel management sites originally considered in this EIS and the spent nuclear fuel distribution alternatives have been kept in the final to maintain maximum consistency with the analyses provided in the Programmatic SNF&INEL EIS (DOE, 1994h and 1995g).

The environmental impacts analyzed pertain to management of foreign research reactor spent nuclear fuel under the basic implementation, and implementation alternatives of Management Alternative 1, the storage of vitrified waste that may be accepted by the United States under Management Alternative 2, and the management of foreign research reactor spent nuclear fuel under Management Alternative 3, the Hybrid Alternative. Chemical separation, which is an implementation alternative to storage, is analyzed in Section 4.3 of this EIS.

Since foreign research reactor spent nuclear fuel is part of the DOE's overall management of spent nuclear fuel, the management options in this EIS must be consistent with the site management alternatives considered in the Programmatic SNF&INEL Final EIS. The alternatives considered in the Programmatic SNF&INEL Final EIS are: Decentralization and 1992/1993 Planning Basis (even distribution of foreign research reactor spent nuclear fuel between the Idaho National Engineering Laboratory and the Savannah River Site), Regionalization by Geography, Regionalization by Fuel Type, and Centralization (all foreign research reactor spent nuclear fuel eligible under the policy).

The foreign research reactor spent nuclear fuel management options also depend on the availability of the sites to implement the policy immediately. Of the five sites, only the Savannah River Site and the Idaho National Engineering Laboratory would be available in late 1995. The other three sites could become available at a later date when appropriate facilities would be completed (either constructed or refurbished). This constraint has necessitated a two-phased approach to foreign research reactor spent nuclear fuel management in the United States in which foreign research reactor spent nuclear fuel is received and managed first at an available management site and is shipped to another site later. For the purpose of this analysis, the implementation of the policy was divided into two functional periods — the period during which receipt and storage of foreign research reactor spent nuclear fuel would be accomplished by using existing facilities (Phase 1), and the period during which new or refurbished facilities could be used (Phase 2). The first phase would be characterized by operational activities only, while the second involves impacts from construction in addition to operational activities.

The environmental impacts from the basic implementation of each Management Alternative, as they relate to storage of the foreign research reactor spent nuclear fuel in the United States, are analyzed in Sections F.4.1 through F.4.5. Elements of this analysis are combined and summarized in Section 4 of this EIS to present the impacts of all Implementation Alternatives under the proposed action.

F.4.1 Savannah River Site

If the Savannah River Site is the site to manage all DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel would be received and managed at the site until ultimate disposition. If the Savannah River Site is not the site to manage DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel could be received and managed at the Savannah River Site until the selected site(s) would be ready to receive the foreign research reactor spent nuclear fuel. The construction of new facilities for managing foreign research reactor spent nuclear fuel is estimated to take about 10 years. Modifications to existing facilities for the same purpose could take less time. This period is referred to as Phase 1. The period following Phase 1 until ultimate disposition is referred to as Phase 2. The amount of spent nuclear fuel that could be received and managed at the Savannah River Site under Management Alternative 1, as discussed in Section 2.2.2, is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly, the Savannah River Site could receive one-half of the foreign research reactor spent nuclear fuel under the Decentralization and the 1992/1993 Planning Basis alternatives, the aluminum-based foreign research reactor spent nuclear fuel under the Regionalization by Fuel Type alternative, the foreign research reactor spent nuclear fuel from eastern ports under the Regionalization by Geography Alternative, or all foreign research reactor spent nuclear fuel under the Centralization Alternative. As discussed in Section 2.6.4.1, the split of foreign research reactor spent nuclear fuel evenly between the Savannah River Site and the Idaho National Engineering Laboratory under the Decentralization and 1992/1993 Planning Basis alternatives in the Programmatic SNF&INEL Final EIS was not considered to have a practical basis, and was therefore not evaluated in detail.

As a potential Phase 1 site under Management Alternative 1, the Savannah River Site would receive and manage foreign research reactor spent nuclear fuel at existing wet storage facilities: RBOF and the L-Reactor disassembly basin. Descriptions of RBOF and the L-Reactor disassembly basin are provided in Section F.3. RBOF is located at the H-Area. It is a facility with provisions for the receipt and storage of irradiated nuclear fuel elements. Since 1963, irradiated spent nuclear fuel elements have been received from offsite reactors and from the Savannah River Site reactors. RBOF provides the capability for underwater unloading of the transportation casks and the handling and storage of the foreign research reactor spent nuclear fuel. The foreign research reactor spent nuclear fuel would be stored in RBOF until its storage capacity is exhausted. Currently, RBOF has space for approximately 1,170 foreign research reactor spent nuclear fuel elements. This capacity could be increased to a total of 2,425 elements by rearrangement and consolidation of existing inventory (O'Rear, 1995).

The L-Reactor disassembly basin is not currently configured for storage of aluminum-based foreign research reactor spent nuclear fuel; however, minor modifications which would provide new storage racks, new handling equipment, safety documentation, etc., along with upgrades in progress to address vulnerabilities associated with water chemistry control, would permit receipt and management of foreign research reactor spent nuclear fuel. Installation of racks equivalent to those in RBOF would provide storage for approximately 20,000 foreign research reactor spent nuclear fuel elements. The modifications to RBOF and L-Reactor disassembly basin are part of the ongoing programs at the site to be performed independent of the proposed action in this EIS.

Between the RBOF and the L-Reactor disassembly basin, there would be sufficient storage capacity and handling capability to accommodate the receipt and management of foreign research reactor spent nuclear fuel during the estimated 10-year period for Phase 1.

An additional option to enhance storage capacity during Phase 1 would be to use RBOF and/or L-Reactor disassembly basin to unload the transportation casks and provide storage capacity in dry storage casks which would be placed near the existing facility. Descriptions of the dry storage casks are provided in Section F.3.

As a Phase 2 site under the basic implementation of Management Alternative 1, the Savannah River Site would continue to receive foreign research reactor spent nuclear fuel beyond Phase 1 in a new dry storage facility that would be constructed at the H-Area. The H-Area is the preferred site among several considered for the construction of new foreign research spent nuclear fuel storage facilities, and is the location assumed for the environmental impacts calculations. An alternative site, equally qualified for construction of new storage facilities is located on a ridge between the P-Reactor and the Pen Branch watershed as indicated in Section 2, Figure 2-14 of this EIS (Shedrow, 1994a). Foreign research reactor spent nuclear fuel stored during Phase 1 would be transferred to the new facility and would be stored there for an additional 30 years until ultimate disposition. The dry storage would encompass a number of designs, examples of which were provided in Section 2.6.5.1.1 and in Section F.3.

The analysis of environmental impacts from the management of foreign research reactor spent nuclear fuel at the Savannah River Site is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set which provides a typical, and in many cases, bounding estimate of the resulting impacts.

The specific analysis options are as follows:

1A. The Savannah River Site would receive foreign research reactor spent nuclear fuel during Phase 1 and store it at the RBOF and/or the L-Reactor disassembly basin. For the purpose of this analysis, the amount of fuel to be stored is all foreign research reactor spent nuclear fuel that would be received during Phase 1 (approximately 17,500 elements). The spent nuclear fuel would be shipped offsite at the end of Phase 1.

1B. Foreign research reactor spent nuclear fuel stored under analysis option 1A would be transferred to a newly constructed dry storage facility, where it would be stored until ultimate disposition. Foreign research reactor spent nuclear fuel arriving in the United States after Phase 1 concludes would be received and stored at the new dry storage facility. For the purpose of this analysis, the amount of spent nuclear fuel that would be stored would be all the foreign research reactor spent nuclear fuel eligible under the policy (22,700 elements).

The implementation alternatives of Management Alternative 1 for managing foreign research reactor spent nuclear fuel in the United States, as discussed in Section 2.2.2, introduce additional analysis options that would be considered for the Savannah River Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Savannah River Site would be likely to receive and manage foreign research reactor spent nuclear fuel in existing facilities during the Phase 1 period. The impacts would be bounded by analysis option 1A (above). Impacts of construction and operation of the dry storage facility considered in analysis option 1B would bound those of the facility required to accommodate this amount of fuel. The spent nuclear fuel would either be shipped offsite after Phase 1, or it would be managed along with the rest of the spent nuclear fuel at the Savannah River Site.

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Savannah River Site would receive only HEU from the foreign research reactors eligible under the policy. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of spent nuclear fuel at the Savannah River Site would be bounded by analysis options 1A and 1B above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Savannah River Site would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which in uranium content represents approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis options 1A or 1B (above) by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years, and therefore the amount of spent nuclear fuel available for acceptance would also be decreased. The impacts from the management of the decreased amount of spent nuclear fuel at the Savannah River Site would be bounded by analysis options 1A or 1B above.

- Under Implementation Subalternative 2b, (Section 2.2.2.2), the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and stored would remain constant. The impacts would be the same as in analysis options 1A or 1B.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be accepted by the United States because the foreign research reactors would consider their own alternatives as to whether or not to send the spent nuclear fuel to the United States. The amount of foreign research reactor spent nuclear fuel, in this case, cannot be quantified. The upper limit, however, as considered under analysis options 1A and 1B (above), would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of foreign research reactor spent nuclear fuel would be taken. The choices do not affect the impacts at the Savannah River Site.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider wet storage technology for new construction. DOE would implement the policy by constructing a new wet storage facility at the H-Area or by using the BNFP, owned by Allied General Nuclear Services. DOE would have to acquire the facility which could be ready for use in approximately 5 years. Therefore, if the Savannah River Site is selected under either the Regionalization or Centralization Alternatives of the Programmatic SNF&INEL Final EIS, Phase 2 at the Savannah River Site could start as early as 5 years from the start of implementation period by using BNFP. The new wet storage facility is described in Section 2.6.5.1.2. BNFP is described in Section F.1. For this implementation alternative, an analysis option 1C is considered, which is similar to 1B, as follows:

1C. The spent nuclear fuel managed under analysis option 1A would be transferred to a newly constructed wet storage facility or the BNFP where it would be managed until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 would be received and managed at these facilities. For the purpose of this analysis, the amount of spent nuclear fuel that would be managed in these facilities would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements).

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. As noted in Section 2.3.6, the Savannah River Site is currently limited to chemical separation of aluminum-based foreign research reactor spent nuclear fuel.

Under Management Alternative 2, as discussed in Section 2.3, DOE and the Department of State would assess the management of foreign research reactor spent nuclear fuel in a foreign location which would include an evaluation of foreign reprocessing with acceptance by the United States of the vitrified high-level waste resulting from reprocessing. The waste would be received and managed at the Defense Waste Process Facility at the Savannah River Site. DOE estimates that the total volume of the vitrified high-level waste would be about 2.4 m^3 (8.5 ft^3) and it would fill about 16 European-size canisters. A European-size canister is about four times smaller than the canister used in the Defense Waste Process Facility at the Savannah River Site.

Under Management Alternative 3 (Hybrid Alternative), as discussed in Section 2.4, the Savannah River Site would receive the aluminum-based fuel which would not be reprocessed overseas. This spent nuclear fuel would be processed at the Savannah River Site chemical separation facilities in the same manner as in Implementation Alternative 6 above. The amount of foreign research reactor aluminum-based spent nuclear fuel to be chemically separated would be approximately 12,200 elements, 12.9 MTHM, 79 m^3 (2,600 ft^3).

F.4.1.1 Existing Facilities (Phase 1)

Analysis option 1A utilizes existing facilities that would be ready to receive and store foreign research reactor spent nuclear fuel by late 1995. The environmental impacts from this analysis option include only those related to operations, specifically: socioeconomics; occupational and public health and safety; materials, utilities, and energy; air quality; and waste management. For this analysis, it was assumed that the amount of foreign research reactor spent nuclear fuel to be received at the management site is the maximum, and the receipt rate is uniform at approximately 1,800 elements per year.

F.4.1.1.1 Socioeconomics

Potential socioeconomic impacts associated with analysis option 1A would be attributable to the staffing requirements for existing facilities. Currently, these facilities are being used to store spent nuclear fuel, so any incremental staffing requirements related to foreign research reactor spent nuclear fuel storage would be small. All personnel required for the operation and support of the existing facilities could be acquired from the current work force at the Savannah River Site. Use of the current work force would not result in any net socioeconomic impact relative to baseline employment data. In fact, using the current work force may partially compensate for the decline in employment expected from changes in site mission from 20,000 persons in 1995 to approximately 15,800 persons in 2004 (DOE, 1995g).

F.4.1.1.2 Occupational and Public Health and Safety

Radiological exposures could affect occupational and public health and safety. Possible sources of radiological exposure from the receipt and storage of foreign research reactor spent nuclear fuel include: (1) airborne emissions from incident-free operations; (2) incident-free handling activities; and (3) airborne emissions from accident conditions. Radiological exposures are presented in individual subsections for emissions-related impacts and handling-related impacts. Accident-related impacts are presented in Section F.4.1.3.

Emissions-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Savannah River Site would be attributed to airborne emissions of radioactive material that could be carried by wind offsite. The general public would be too far from the locations where handling activities or storage would take place to receive any dose from direct exposure. Doses were calculated for the maximally exposed individual (MEI), defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the existing storage facility (RBOF and/or L-Reactor disassembly basin) during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-21 summarizes the annual emission-related doses to the public and the associated risks for the MEI and the population at the Savannah River Site during Phase 1 operations.

Table F-21 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage in Existing Facilities at the Savannah River Site (Phase 1)

	MEI Dose (mrem/yr)	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• RBOF (wet)	0.00011	5.5×10^{-11}	0.0057	0.0000028
• L-Reactor Basin (wet)	0.000073	3.7×10^{-11}	0.0046	0.0000023
Storage at:				
• RBOF (wet)	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
• L-Reactor Basin (wet)[a]	0.00036	1.8×10^{-10}	0.022	0.000011

[a] *L-Reactor basin doses are due to existing conditions. The foreign research reactor spent nuclear fuel contribution would be six orders of magnitude lower.*

Handling-Related Impacts: Management site workers would receive radiation doses during handling operations, such as receiving and unloading the transportation casks, transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 1A involves the receipt of 644 shipments of spent nuclear fuel into the existing wet storage facility (RBOF and/or L-Reactor disassembly basin) during Phase 1, and the preparation of 161 transportation casks for offsite shipment at the end of Phase 1. It was assumed that at the end of a 10-year period (i.e., Phase 1), the spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. The assumptions and methodology used to calculate the doses to a working crew associated with the handling activities of the spent nuclear fuel are described in Section F.5 of this appendix.

The collective doses that would be received by the members of the working crew and the associated risk were calculated for Phase 1 operations. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculation. However, the upper bound for such a dose would be equal to administrative or regulatory limits at the management site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If one worker received the full 5,000 mrem per year for the full 13 years of potential spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. The associated risk of incurring a latent cancer fatality (LCF) would be 2.6 percent. The collective dose to the workers would be 250 person-rem with an associated LCF risk of 0.10.

F.4.1.1.3 Material, Utility, and Energy Requirements

The estimated annual consumption of materials, utilities, and energy from the use of existing storage facilities is shown in Table F-22.

Table F-22 Annual Utility and Energy Requirements for Foreign Research Reactor Spent Nuclear Fuel Storage at Existing Facilities at the Savannah River Site (Phase 1)

Commodity	*Baseline Site Usage*	*RBOF*	*L-Reactor Basin*	*Percent Increase*
Electricity (MW-hr/yr)	659,000	1,430	784	0 percent
Fuel (l/yr)	28,400,000	6,570	15,000	0.05 percent
Water (l/yr)	88,200,000,000	35,100,000	2,900,000	0.04 percent

The material, utility, and energy requirements for analysis option 1A would represent a small percentage of current requirements. No new generation or treatment facilities would be necessary. Increases in fuel consumption at the Savannah River Site would be minimal because overall onsite activity would not increase due to changes in the Savannah River Site mission and the general reduction in employment levels. The Programmatic SNF&INEL Final EIS concluded that the existing capacities and distribution systems for electricity, steam, water, and domestic wastewater treatment are adequate to support any of the five alternatives considered for spent nuclear fuel management at the Savannah River Site. This conclusion would also be valid for analysis option 1A because it is bounded by the alternatives considered in the Programmatic SNF&INEL Final EIS (DOE, 1995g).

F.4.1.1.4 Waste Management

The estimated annual waste generation for foreign research reactor spent nuclear fuel at the RBOF and L-Reactor disassembly basin is shown in Table F-23. These quantities represent a very small percent increase above current levels at the Savannah River Site. Existing waste management storage and disposal activities at the Savannah River Site could accommodate the waste generated by foreign research reactor spent nuclear fuel storage at the RBOF and L-Reactor disassembly basin. Therefore, the impact of this waste on existing Savannah River Site waste management activities would be minimal.

Table F-23 Annual Waste Generation for Foreign Research Reactor Spent Nuclear Fuel Storage at Existing Facilities at the Savannah River Site (Phase 1)

Waste Form	*Baseline Site Generation*	*RBOF*	*L-Reactor Basin*	*Percent Increase*
High-Level Waste (m^3/yr)	127,400[a]	0	0	0 percent
Transuranic Waste (m^3/yr)	760	0	0	0 percent
Solid Low-Level Waste (m^3/yr)	19,750	161	510	2.6 percent
Wastewater (l/yr)	690,000,000	2,650,000	35,000,000	5.1 percent

[a] *Total inventory (m^3) at the Savannah River Site.*

F.4.1.1.5 Air Quality

Nonradiological Emissions: Impact assessments for nonradiological air emissions associated with implementation of the respective spent nuclear fuel management alternatives (excluding construction-related activities) are based primarily on analyses performed for the Programmatic SNF&INEL Final EIS (DOE, 1995g). These analyses were based on the following assumptions and qualifications:

- Air emissions data for wet storage are based upon releases from the RBOF
- All air pollutant sources, except standby diesel generators, are assumed to operate continuously. The standby diesel generators are assumed to operate daily for 1 hour; whereas their actual operation consists of a single monthly test.

Table F-24 lists the annual potential maximum emissions of criteria and toxic air pollutants (in tons per year) attributable to existing facilities. These data indicate little or no difference in pollutant loading between the baseline and the regionalization alternative considered in the Programmatic SNF&INEL Final EIS. The greatest pollutant contribution for criteria air pollutants would be for nitrogen oxides (7.7 metric tons per year, or 8.5 tons per year) and carbon monoxide (1.8 metric tons per year, or 2.0 tons per year). Assuming that the foreign research reactor spent nuclear fuel comprises approximately nine percent of the total spent nuclear fuel managed under the Centralization Alternative of the Programmatic SNF&INEL Final EIS, the incremental and cumulative nonradiological air quality impacts attributable to the storage of foreign research reactor spent nuclear fuel in existing facilities (RBOF, L-Reactor disassembly basin) would be small.

Table F-24 Annual Maximum Emissions of Criteria Air Pollutants Attributable to Foreign Research Reactor Spent Nuclear Fuel Storage at Existing Facilities at the Savannah River Site (Phase 1)

Pollutant	*Baseline Wet (tons/yr)*[a,b,c]	*RBOF (tons/yr)*[a,b]	*Percent Increase*
Sulfur Oxides	0.4	0.005	1.3 percent
Nitrogen Oxides	6.0	0.77	12.8 percent
Total Suspended Particulates	0.4	0.006	1.5 percent
Carbon Monoxide	1.5	0.18	12 percent
Total Volatile Organic Compounds	0.6	0.077	12.8 percent
Gaseous Fluorides	none	none	none

[a] *Source: Hunter and Stewart, 1994*

[b] *To convert tons to metric tons, multiply by 0.907*

[c] *Decentralization based on management of the existing Savannah River Site inventory of spent nuclear fuel*

Radiological Emissions: The ventilation system serving the RBOF is designed to minimize airborne radioactivity levels both inside and outside of the facility. This ventilation system is based on a "once through" multiple-air-zone concept in which air flows from areas with low potential for contamination to areas with higher potential. The airflow is passed through High-Efficiency Particulate Air filters mounted on the building exhaust. A review of 1993 emissions data from the RBOF indicates emissions of approximately 2.7×10^{-7} Ci per year of ^{137}Cs (DOE, 1995g).

The RBOF and L-Reactor disassembly basin are currently being utilized to wet store spent nuclear fuel, and their emissions are reflected within baseline environmental conditions.